~~**Proprietary Information - Withhold From Public Disclosure Under 10 CFR 2.390**~~
The balance of this letter may be considered non-proprietary upon removal of Attachment 9.

Entergy

Entergy Operations, Inc.
17265 River Road Hwy 18
Killona, LA 70057
Tel 504-739-6660

Michael R. Chisum
Site Vice President
Waterford 3

W3F1-2015-0040

July 2, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request to Revise Control Element Assembly Drop Times Associated with Technical Specification 3.1.3.4
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy Pre-submittal Meeting Summary for the Revised Control Element Assembly Drop Times [ADAMS Accession Number ML15117A503].
2. Entergy Pre-Submittal Meeting Revised Presentation Slides for Control Element Assembly Drop Times [ADAMS Accession Number ML15113A787].

Dear Sir or Madam:

On April 22, 2015, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of Entergy Operations, Inc. (Entergy) and Westinghouse Electric Company (Westinghouse) at the NRC Headquarters. The purpose of the meeting was to discuss the Entergy's license amendment request (LAR) regarding changes to Technical Specification (TS) 3.1.3.4 (CEA Drop Time) and Final Safety Analysis Report (FSAR) Chapter 15 (Accident Analyses). Reference 1 provides the meeting summary information and Reference 2 provides the meeting presentation information.

As discussed in the public meeting and pursuant to 10 CFR 50.90, Entergy hereby requests an amendment to revise the Control Element Assembly (CEA) drop times associated with Technical Specification 3.1.3.4 for Waterford Steam Electric Station Unit 3 (Waterford 3).

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using the criteria in 10 CFR 50.92(c), and it has been determined that the changes involve no

significant hazards consideration. The bases for these determinations are included in Attachment 1. The proposed change includes four new commitments (Attachment 7).

Attachment 9 is proprietary in its entirety, as it contains information that is proprietary to Westinghouse Electric Company (Westinghouse). Attachment 8 contains the Proprietary Information Affidavit. The purpose of this attachment is to withhold the proprietary information contained in Attachment 9 from public disclosure. The Affidavit, signed by Westinghouse as the owner of the information, sets forth the basis for which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of § 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR 2.390.

Entergy requests approval of the proposed amendment by October 25, 2015. Once approved, the amendment shall be implemented within 60 days.

If you have any questions or require additional information, please contact John Jarrell, Regulatory Assurance Manager, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 2, 2015.

Sincerely,

MRC/JPJ/wjs

Attachments:

1. Analysis of Proposed Technical Specification Change
2. Final Safety Analysis Report Revised Assessment
3. Final Safety Analysis Report Revised Assessment Summary
4. Average Control Element Assembly Drop Time Basis
5. Revised (Markup) Technical Specification Pages
6. Clean (Revised) Technical Specification Pages
7. List of Regulatory Commitments
8. Fuel Thermal Conductivity Degradation Evaluation Proprietary Affidavit
9. **PROPRIETARY** - Fuel Thermal Conductivity Degradation Evaluation

cc: Mr. Marc L. Dapas
Regional Administrator
U. S. NRC, Region IV
RidsRgn4MailCenter@nrc.gov

NRC Senior Resident Inspector for Waterford 3
Frances.Ramirez@nrc.gov (SRI)
Chris.Speer@nrc.gov (RI)

NRC Program Manager for Waterford 3
Michael.Orenak@nrc.gov

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
Ji.Wiley@LA.gov

Attachment 1 to

W3F1-2015-0040

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

On April 22, 2015, a Category 1 public meeting was held between the U.S. Nuclear Regulatory Commission (NRC) staff and representatives of Entergy Operations, Inc. (Entergy) and Westinghouse Electric Company (Westinghouse) at the NRC Headquarters. The purpose of the meeting was to discuss the Entergy's license amendment request (LAR) regarding changes to Technical Specification (TS) 3.1.3.4 (CEA Drop Time) and Final Safety Analysis Report (FSAR) Chapter 15 (Accident Analyses). Reference 7.30 provides the meeting summary information and Reference 7.31 provides the meeting presentation information.

As discussed in the public meeting and pursuant to 10 CFR 50.90, Entergy hereby requests an amendment to revise the Control Element Assembly (CEA) drop times associated with Technical Specification 3.1.3.4 for Waterford Steam Electric Station Unit 3 (Waterford 3).

As described in Regulatory Guide 1.181 [Reference 7.22], the terminology for "updated FSARs" varies throughout the industry. The terms updated FSAR and UFSAR are equivalent and have the same meanings. In addition, the terms "CEA drop time," "rod insertion time," "CEA SCRAM insertion curve," and "scram time" are equivalent and have the same meanings.

2.0 PROPOSED CHANGE

The proposed change to Waterford 3 TS 3.1.3.4 revises the following:

- The arithmetic average of all CEA Drop Times is revised to be ≤ 3.2 seconds
- The Individual CEA drop times are revised to ≤ 3.5 seconds

The revised TS 3.1.3.4 page is included in Attachment 5. The clean TS 3.1.3.4 page is included in Attachment 6.

3.0 BACKGROUND

The TS 3.1.3.4 required CEA drop time measurement is performed at the beginning of every cycle to ensure the CEAs adequately trip when required. The CEA drop time testing is performed using procedure NE-002-020 (CEA Insertion Time Measurement) [Reference 7.38]. The current TS 3.1.3.4 CEA drop time limits were approved by the NRC in TS Amendment 58 [Reference 7.33]. The average drop time limit submittal [Reference 7.32] justified that negative reactivity insertion for any reasonable distribution of CEAs is more directly correlated to the average CEA insertion rather than by the slowest. Attachment 4 validates that the average CEA drop time methodology remains applicable to the new CEA drop times.

The CEA drop time surveillance for Cycle 20, Waterford 3 experienced a challenge in meeting the requirements of TS 3.1.3.4 governing CEA drop time testing. TS 3.1.3.4 states:

> The arithmetic average of the CEA drop times of all CEAs from a fully withdrawn position, shall be less than or equal to 3.0 seconds; and the individual CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.2 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with average reactor coolant system temperature greater than or equal to 520°F and all reactor coolant pumps operating.

Historically, Waterford 3 had not been challenged to pass the CEA drop time test. The apparent cause evaluation identified the most probable cause as being the combined effects of major plant modifications that increased the resistance experienced by the CEAs during the insertion from 100% withdrawn to 90% inserted. The apparent cause analysis also identified two potential causes.

Apparent Cause: The apparent cause of the rise in CEA drop times is the combined effects of major plant modifications that increased the resistance experienced by the CEAs during the insertion from 100% withdrawn to 90% inserted.

Examples:

1. Increased pressure drop through the fuel assemblies as a result of the transition to NGF Fuel Assemblies
2. Increased core flow as a result of increased cross-sectional area of the RSG U-tubes
3. Lower pressure drop in the RSGs that results in increased flow through the CEA guide tubes

Potential Causes: The first potential cause is the potential rise in CEA drop times due to a change in Control Element Drive Mechanisms (CEDM) voltage decay time between the original and replacement CEDMs causing a delay in the start of the CEA drop. An action was issued to evaluate performing CEDM holding coil decay time testing during Cycle 21 to determine if the actual voltage decay time remains less than the 0.6 seconds assumed in the analysis. This measurement would confirm that the analysis assumptions are correct. Attachment 7 commitment is included to perform this testing.

In addition, Table 4.0-1 (CEA Drop Times) provides the revised CEA drop time curve used in the updated evaluations. The potential CEA holding coil delay could change this curve such that the CEAs may begin inserting at a later time. The limiting analyses will be re-evaluated with a CEA holding coil delay of 0.8 seconds to determine the impact and will be sent as a supplement to this letter. The performance of the revised CEA holding coil decay analysis is included in Attachment 7 as a commitment. This potential cause was identified after the Westinghouse evaluation had already

implemented the Table 4.0-1 curve. The supplement will provide a bounding analysis to ensure conservative results for the limiting events.

A second potential cause could be the rise in CEA drop times due to the change in as-built manufacturing tolerances in the replacement reactor pressure vessel head (RPVH) resulting in increased friction between the CEA extension shaft and the RPVH and thus slower CEA motion.

It was determined that the apparent and potential causes are due to one-time plant modifications and therefore are not expected to further degrade. Since, no further degradation is expected, the action of raising the CEA drop time limit will resolve the lack of CEA drop time margin. Waterford 3 recognizes that confirmation of no further degradation would benefit the NRC review of this submittal, so Attachment 7 (Commitments) has added a commitment to provide the NRC with the CEA drop time results from the next TS 3.1.3.4 surveillance. The intent of this commitment is to provide validation that the CEA drop time has remained consistent with the Cycle 19 and Cycle 20 test results.

4.0 TECHNICAL ANALYSIS

The requirements of TS 3.1.3.4 ensure that actual drop times for CEAs are conservative with respect to the drop time assumed in accident and transient analyses. Specifically, TS 3.1.3.4 currently requires that the average CEA drop time from the fully withdrawn position be less than or equal 3.0 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position, when measured with the average reactor coolant temperature greater than or equal to 520°F with all reactor coolant pumps running. Measurement of CEA drop times under these conditions ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions. CEA drop times are measured following each removal and reinstallation of the reactor vessel head, following any maintenance on, or modification to, the CEA drive system which could affect the drop time of those specific CEAs, and at least once per refueling outage. Should the average CEA drop time be found to exceed 3.0 seconds or an individual CEA exceed 3.2 seconds, the plant cannot proceed to Mode 1 or 2.

4.1 ACCIDENT ANALYSES

The Waterford 3 design basis transient and accident analyses have been reviewed relative to the FSAR to assess the impact of the proposed CEA drop times. The assessment focused on the potential impacts of increased CEA drop time, the accidents can be grouped into four basic categories.

Key to Categories

1. Reactor trip does not occur or is not credited.
2. Consequences are not sensitive to the CEA drop time change, because of the slow rate of margin degradation through the time of trip, or due to obvious insensitivity of accident consequences as a function of the time of trip.
3. This event is bounded by another event that is presented in Chapter 15.
4. This event is evaluated for CEA drop time change impact.

Attachment 2 (Final Safety Analysis Report Assessment) provides the detailed FSAR transient and accident evaluation. Attachment 3 (Final Safety Analysis Report Assessment Summary) provides a summary table and their corresponding category.

The assessment begins with each Analysis of Record (AOR). The current AOR transient and accident analyses are presented in the Waterford 3 FSAR [Reference 2]. No changes are made to the analysis methodologies or FSAR Topical Report restrictions. Table 4.0-1 provides a comparison of the current and revised CEA drop times. The effects of a longer CEA drop time is accommodated by demonstrating that the affected analyses remain within their applicable acceptance limits or existing conservatisms in the analyses are used to demonstrate acceptance limits are met. Each of the design basis events designated as a Category 4 event are provided with a detailed event evaluation in Attachment 2. The Westinghouse reload analysis methodology has been applied to these Waterford 3 changes.

Table 4.0-1. CEA Drop Times

CEA Insertion (%)	Time (Seconds)	Revised Time (Seconds)
0	0.00	**0.00**
0	0.60	**0.60**
5	0.80	**0.95**
10	0.95	**1.15**
20	1.25	**1.45**
30	1.55	**1.75**
40	1.80	**2.00**
50	2.05	**2.25**
60	2.3	**2.50**
70	2.535	**2.75**
80	2.75	**2.95**
90	3.0	**3.20**
100	3.5	**3.50**

The following events were evaluated for the CEA drop time change. These events started with the AOR and only revised the CEA drop time (reactivity). These analyses demonstrated the results remained within the acceptance limits.

- FSAR 15.1.3.1 Steam System Piping Failures Post-Trip Return-To-Power
- FSAR 15.1.3.3 Steam System Piping Failures: Pre-Trip Power Excursion Analysis
- FSAR 15.2.1.3 Loss of Condenser Vacuum
- FSAR 15.2.2.5 Loss of Normal Feedwater Flow
- FSAR 15.2.3.1 Feedwater System Pipe Breaks
- FSAR 15.2.3.2 Loss of Normal Feedwater Flow
- FSAR 15.4.1.1 Uncontrolled CEA Withdrawal from a Subcritical
- FSAR 15.4.1.2 Uncontrolled CEA Withdrawal at Low Power
- FSAR 15.4.3.2 CEA Ejection

The following events were evaluated for the CEA drop time change. These events started with the AOR and used existing conservatisms in the analyses to demonstrate the results remained within the acceptance limits.

- FSAR 15.1.2.3 Increased Main Steam Flow
- FSAR 15.1.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve
- FSAR 15.3.2.1 Total Loss of Forced Reactor Coolant Flow
- FSAR 15.3.3.1 Single Reactor Coolant Pump (RCP) Shaft Seizure / Sheared Shaft
- FSAR 15.9.1 Asymmetric Steam Generator Transient

For the events which took credit for existing analysis margin, changes from the AOR to the revised analysis start with the revised CEA drop time (reactivity), then the following analysis margins were utilized when needed. Attachment 2 provides the details on the event specific margins utilized.

- Axial shape index (ASI) limits were reduced from an ASI of +0.3 to +0.2 which is conservative with respect to the COLR Section 3.2.7 limit for power greater than 50%.
- Least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{^{o}K}$. This still includes the UFSAR 15.0.3.3.1 uncertainty multiplier of 0.85 to conservatively minimize the analysis Doppler impact. The reduction in least negative Doppler margin had an insignificant impact on the minimum DNBR.
- Utilizing the Core Operating Limit Supervisor System (COLSS) initial thermal margin. No changes to COLSS were required. For the analyses that used additional initial thermal margin, COLSS already had the additional thermal margin preserved but it had not been utilized in the AOR.

The Attachment 2 results show that all FSAR transient and accident analyses remain within their acceptance criteria for peak primary pressure, peak secondary pressure,

DNBR, peak linear heat rate, fuel centerline temperature, operating margins, and fuel failure limits. Thus, the revised CEA drop times are acceptable.

The radiological consequences were not adversely impacted by this change. The Attachment 2 Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to all the events. The analysis showed that the differences in primary and secondary system energy after reactor trip are insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. In addition, the Attachment 2 results for each of the individual events demonstrate that the fuel failure limits remain unchanged which mean the radiological source terms remain the same. Thus, there is no change to the radiological results.

In order to aid the NRC review, the relevant Waterford 3 licensing basis history is provided. The last major plant change that submitted the accident and transient analyses to the NRC for review was the extended power uprate [Reference 7.15] and alternate source term implementation [Reference 7.18]. Two additional major plant changes that have been implemented since the extended power uprate are the use of Next Generation Fuel (NGF) and Steam Generator replacement. These changes were addressed by NRC approvals (Table 4.0-2) and the 10CFR50.59 process. The Westinghouse reload analysis methodology has been applied to implement these Waterford 3 changes.

Table 4.0-2 (NRC Amendments of Interest) lists the NRC Amendments that Waterford 3 considered of interest to the NRC to aid in submittal review. The Table 4.0-2 list is not all inclusive but contains the amendments Waterford 3 considered to have an impact on the accident and transient analyses.

Table 4.0-2. NRC Amendments of Interest

NRC Amendment	Date	Summary
NUREG-0787	1981-1985	**Original Licensing** The original licensed power for Waterford 3 was 3390 MWt. The accident and transient analyses were originally approved by the NRC in NUREG-0787 and its supplements [Reference 7.3 through 7.11].
183	2002	**Appendix K Power Uprate** In Waterford 3 letter W3F1-2001-0091 [Reference 7.12], Waterford 3 requested that the Operating License be amended to reflect an increase in the licensed reactor power level from 3,390 MWt to 3,441 MWt utilizing the new 10CFR50 Appendix K rule requirements. The Waterford 3 power measurement uncertainty was reduced by increasing the feedwater flow measurement

NRC Amendment	Date	Summary
		accuracy by utilizing high accuracy ultrasonic flow measurement instrumentation. The NRC approved the Waterford 3 request in NRC Amendment 183 [Reference 7.13].
198	2005	**Alternate Source Term** In Waterford 3 letter W3F1-2004-0053 [Reference 7.16], Waterford requested to use of alternative source term methodology for the determination of radiological consequences described in Regulatory Guide 1.183 [Reference 7.17]. The NRC approved the Waterford 3 use of the alternative source term methodology in NRC Amendment 198 [Reference 7.18].
199	2005	**Extended Power Uprate** In Waterford 3 letter W3F1-2003-0074 [Reference 7.14], Waterford 3 requested a power increase associated with an extended power uprate. For Cycle 14, Waterford 3 implemented an extended power uprate to a licensed thermal power level of 3716 MWt. All of the Chapter 15 design basis events were re-assessed for the extended power uprate. The Westinghouse reload analysis methodology was applied to determine the impact of the extended power uprate on Waterford 3. The extended power uprate was approved in NRC Amendment 199 [Reference 7.15].
200 210 214 215 224	2005-2009	**Next Generation Fuel** The implementation of next generation fuel with changes in fuel poisons were approved by the NRC in multiple amendments. The NRC Amendments 200 [Reference 7.19], 210 [Reference 7.20], 214 [Reference 7.21 and 7.22], 215 [Reference 7.23 and 7.24], and 224 [Reference 7.25] apply to different parts of the new fuel designs and implementation.
225 231 232 236	2012	**Replacement Steam Generator and Reactor Head** The outage prior to Cycle 19, Waterford 3 performed a replacement steam generator and reactor head outage. The NRC Amendments applicable to this change were 225 [Reference 7.26], 231 [Reference 7.27], 232 [Reference 7.28], and 236 [Reference 7.29] and apply to different parts of the change.

4.2 Fuel Thermal Conductivity Degradation Evaluation

The FATES3B fuel performance code is used to calculate fuel temperatures and rod internal pressures for the safety analysis and thermal-mechanical calculations for CE-NSSS plants. The FATES3B code was developed to conservatively bound the fuel temperature and fission gas release data measured, and consequently provide a conservative prediction of rod internal pressure. In addition to the conservatisms built into the models, the application methodology of the FATES3B code, especially through

the use of bounding power histories, bounding axial power shapes, and accounting for peaking factor reduction with burnup compensates for the effects of thermal conductivity degradation for the safety and design applications.

The margins listed above require significant time and effort to quantify, and would require even more effort to review. In order to expedite the licensing process, the Waterford Unit 3 core design will retain margin to the radial fall-off (RFO) used in non-LOCA and LOCA safety analyses. The same approach has been used previously for another CE-NSSS plant [Reference 7.39]. Attachment 9 contains the Westinghouse proprietary evaluation provided for Waterford 3 with respect to fuel thermal conductivity degradation. Attachment 8 is the Westinghouse fuel thermal conductivity degradation proprietary affidavit.

Attachment 7 provides the following commitment which is consistent with that implemented for another CE-NSSS plant [Reference 7.24]:
The radial power fall-off curve limits shall be verified each cycle as part of the Westinghouse reload analysis methodology until a new licensing basis long term fuel methodology is approved Waterford 3. Upon NRC approval of a new long term fuel evaluation model and associated methods that explicitly account for thermal conductivity degradation (TCD) that is applicable to Waterford Unit 3 design, Entergy will, within 6 months:

a. Demonstrate that Waterford Unit 3 safety analysis remain conservatively bounded in licensing basis analyses when compared to the NRC-approved new long term fuel evaluation model that is applicable to Waterford Unit 3 design, and/or
b. Provide a schedule for re-analysis using the NRC-approved new long term fuel evaluation model that is applicable to Waterford 3 design for any affected licensing basis analyses.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. The control element assemblies comply with 10CFR50 Appendix A, General Design Criteria (GDC) related to equipment safety functions (GDCs 1, 2, 14, 29), suppression of reactor power oscillations (GDC 12), instrumentation and control equipment (GDC 13), protection systems (GDCs 20, 23, 25), and reactivity control systems (GDCs 10, 26, 27, 28). Negative reactivity insertion rates relate to requirements of 10 CFR 50.36 Technical Specification. As discussed below, the change provided in this TS amendment does not affect the conclusions provided in the UFSAR, and the CEAs and related systems continue to comply with the regulation.

GDCs 1, 2, 14, 29 - Equipment safety functions: These GDCs, as they relate to the CEA system, require that the CEAs be designed to quality standards commensurate with the importance of the safety functions in the event of anticipated operational occurrences (AOOs); be designed to withstand the effects of an earthquake without loss of capability to perform its safety functions; the reactor coolant pressure boundary (RCPB) portion of the CEA system shall be designed, constructed, and tested for the extremely low probability of leakage or gross rupture; and the CEAs shall be designed to assure an extremely high probability of accomplishing their safety functions. As the only parameter that is being changed in this LAR is the CEA drop time, none of the above requirements are being affected.

GDC 12 - Reactor power oscillations: This GDC requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. The CEAs will continue to provide fundamental mode stability even after this TS change. Likewise, there is no change to the ability of the CEAs to suppress the axial mode oscillations.

GDC 13 - Instrumentation and control equipment: This GDC requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. The control of reactor power by means of CEAs is provided (among other systems) to monitor and maintain control over the fission process during both transient and steady state periods over the lifetime of the core. The control by means of CEAs is not affected by the increase CEA drop time.

GDCs 20, 23, 25 - Protection systems: These GDCs require that the plant protection system be designed: (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs, (2) to sense accident conditions and to initiate the operation of systems and components important to safety, (3) to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis, and (4) to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal of control rods. In order to support these GDCs, a reactor trip is accomplished by de-energizing the control element drive mechanism (CEDM) holding coils through the interruption of the CEDM power supply. The CEAs are released to drop into the core reducing reactor power. The accident and transient analysis rate of negative reactivity insertion with the propose change ensures the event consequences remains below the regulatory acceptance limits. The above functions continue to be available even with the change in TS provided in this LAR.

GDCs 10, 26, 27, 28 - Reactivity control systems: These GDCs require that one of two reactivity control systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, specified acceptable safety fuel design limits (SFDL) are not exceeded. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to shutdown the reactor and cool the core is maintained. The reactivity control systems shall also be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.

The CEAs are inserted into the reactor core by a positive means (gravity) and they are capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including specified AOOs, specified acceptable fuel design limits are not exceeded. The CEAs can also be mechanically driven into the core. The appropriate margin for a stuck rod is provided by assuming in the analyses of AOOs that the highest worth CEA does not fall into the core. In addition, the Safety Injection System, in conjunction with the combined capabilities of the reactivity control systems is available to maintain short and long term cooling of the core even in the event a CEA of highest worth is stuck out of the core. None of the above functions of the CEAs are altered or affected by the TS change provided in this LAR.

In conclusion, on the basis of the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.2 No Significant Hazards Consideration

The proposed change will modify Waterford 3 Technical Specification (TS) 3.1.3.4 related to the Control Element Assembly (CEA) drop time limits. Waterford 3 has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, “Issuance of amendment,” as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

The proposed change increases the TS 3.1.3.4 average and individual CEA drop time limits. The CEA drop time is required to be verified prior to Modes 1 or 2 of plant operations. The probability of an accident previously evaluated remains unchanged since the CEAs drop into the core as a result of an accident or transient condition, and the fact that the CEA drop time was increased does not in itself initiate an accident.

The proposed change to the CEA drop time requirements have been evaluated for impact on the accident analyses. The accident analyses all remain within the regulatory acceptance criteria.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change increases the TS 3.1.3.4 average and individual CEA drop time limits. The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing plant operations. The proposed change will not introduce new failure modes or effects and will not, in the absence of other unrelated failures, lead to an accident whose consequences exceed the consequences of accidents previously analyzed.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The increase in CEA drop time as proposed in this TS 3.1.3.4 change has been determined to have no significant impact on the accident analyses described in the FSAR which means this change does not have a significant reduction on the existing margins of safety for the fuel, the fuel cladding, the reactor coolant system boundary, or the containment building. The change in CEA drop time does not impact the fuel rod design or mechanical design analysis. The slightly slower drop time would produce a smaller impact on the fuel assembly and lower stresses on the CEAs. The accident analysis consequences became slightly

more adverse but all remained within the regulatory acceptance limits, thus this change does not involve a significant reduction in a margin of safety.

Based on this analysis, it was determined that the proposed amendment does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any previously evaluated; nor (3) involve a significant reduction in a margin of safety. Therefore, the amendment does not involve a significant hazards consideration.

5.3 Environmental Considerations

The proposed amendment does not change any requirements with respect to the installation of or use of a facility component located within the restricted area, as defined in 10CFR20, or change any inspection or surveillance requirement. The proposed amendment does not involve (i) a significant hazards consideration,
(ii) a significant change in the types or significant increase in the amount of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

Waterford 3 previously requested a Technical Specification change to the CEA drop methodology in letter W3P89-3094 [Reference 7.37]. This submittal was approved in NRC Technical Specification Amendment 58 [Reference 7.38]. This change request is similar in scope to that previously requested and approved.

Arkansas Unit 2 requested a Technical Specification change to increase the individual CEA drop time in letter 2CAN080701 [Reference 7.39]. The NRC approved this request in NRC Technical Specification Amendment 275 [Reference 7.40]. This change was necessitated by the transition to Next Generation Fuel similar to Waterford 3's submittal request.

St. Lucie Unit 2 requested a Technical Specification change to increase the CEA drop time in letter L-2009-127 [Reference 7.41]. The NRC approved this request in NRC Technical Specification Amendment 158 [Reference 7.42].

7.0 REFERENCES

7.1 Waterford Nuclear Generator Station Unit 3, Technical Specifications, Through Amendment 242.

7.2 Waterford Nuclear Generator Station Unit 3, Updated Final Safety Analysis Report (UFSAR), Revision 308 [Letter W3F1-2014-0062 November 11, 2014].

7.3 NUREG-0787, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, July 1981.

7.4 NUREG-0787 Supplement 1, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, October 1981.

7.5 NUREG-0787 Supplement 2, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, January 1982.

7.6 NUREG-0787 Supplement 3, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, April 1982.

7.7 NUREG-0787 Supplement 4, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, October 1982.

7.8 NUREG-0787 Supplement 5, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, June 1983.

7.9 NUREG-0787 Supplement 6, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, June 1984.

7.10 NUREG-0787 Supplement 8, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, December 1984.

7.11 NUREG-0787 Supplement 10, Waterford Steam Electric Station Unit 3, Safety Evaluation Report, March 1985.

7.12 W3F1-2001-0091, Waterford 3 Appendix K Margin Recovery - Power Uprate Request, September 21, 2001 [NRC ADAMS Accession Number ML012700104].

7.13 NRC Issuance of Amendment 183 for the Waterford 3 Appendix K Margin Recovery - Power Uprate, March 29, 2002 [NRC ADAMS Accession Number ML020910734].

7.14 W3F1-2004-0053, Alternate Source Term License Amendment Request, July 15, 2004 [NRC ADAMS Accession Number ML042020294].

7.15 Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.

7.16 NRC Safety Evaluation Report, Waterford 3 Amendment 198, Full Scope Implementation of an Alternative Accident Source Term, March 29, 2005 [NRC ADAMS Accession Number ML050890248].

7.17 W3F1-2003-0074, Extended Power Uprate License Amendment Request, November 13, 2003 [NRC ADAMS Accession Number ML040260317].

7.18 NRC Safety Evaluation Report, Waterford Steam Electric Station, Unit 3 – Issuance of Amendment 199 RE: Extended Power Uprate, April 15, 2005 [NRC ADAMS Accession Number ML051030082].

7.19 NRC Safety Evaluation Report, Waterford 3 Amendment 200, Modification of Technical Specification TS 5.3.1, Fuel Assemblies, TS 5.6.1, Criticality, TS 6.9.1.11.1 Core Operating Limits Reports, and Deletion of TS Index, May 9, 2005 [NRC ADAMS Accession Number ML051290368].

7.20 NRC Safety Evaluation Report, Waterford 3 Amendment 210, Core Operating Limits Report, October 6, 2006 [NRC ADAMS Accession Number ML061930421].

7.21 NRC Safety Evaluation Report, Waterford 3 Amendment 214, Next Generation Fuel, April 15, 2008 [NRC ADAMS Accession Number ML080880014].

7.22 Regulatory Guide 1.181, Content of Updated Final Safety Analysis Report in Accordance with 10CFR50.71(e), September 1999.

7.23 NRC Safety Evaluation Report, Waterford 3 Amendment 215, Optimized Zirlo Fuel Rod Cladding, April 16, 2008 [NRC ADAMS Accession Number ML080380005].

7.24 NRC Safety Evaluation Report, St. Lucie Plant Unit 2 Issuance of Amendment Regarding Extended Power Uprate, September 24, 2012 [NRC ADAMS Accession Number ML12235A463].

7.25 NRC Safety Evaluation Report, Waterford 3 Amendment 224, Revision to the Departure from Nucleate Boiling Ratio Safety Limit, November 3, 2009 [NRC ADAMS Accession Number ML092880237].

7.26 NRC Safety Evaluation Report, Waterford 3 Amendment 225, Technical Specification Change to Relocate Steam Generator High Level Trip Function, March 18, 2010 [NRC ADAMS Accession Number ML093480507].

7.27 NRC Safety Evaluation Report, Waterford 3 Amendment 231, Technical Specification Change Regarding Containment Building Penetrations During Refueling Operations, February 23, 2011 [NRC ADAMS Accession Number ML110140026].

7.28 NRC Safety Evaluation Report, Waterford 3 Amendment 232, Approval of Leak-Before-Break of the Pressurizer Surge Line, February 28, 2011 [NRC ADAMS Accession Number ML110410119].

7.29 NRC Safety Evaluation Report, Waterford 3 Amendment 236, Technical Specification Change Regarding Steam Generator Tube Integrity, July 31, 2012 [NRC ADAMS Accession Number ML12152A341].

7.30 Entergy Pre-submittal Meeting Summary for the Revised Control Element Assembly Drop Times [ADAMS Accession Number ML15117A503].

7.31 Entergy Pre-Submittal Meeting Revised Presentation Slides for Control Element Assembly Drop Times [ADAMS Accession Number ML15113A787].

7.32 Waterford 3 Letter W3P89-3094, Technical Specification Change Request, Average Control Element Assembly Drop Time, August 14, 1989.
7.33 NRC Waterford 3 Technical Specification Amendment 58, Average Control Element Assembly Drop Time, October 31, 1989 [ADAMS Accession Number ML021760257].
7.34 Arkansas Unit 2 Letter 2CAN080701, Technical Specification 3.1.3.4 CEA Drop Time, August 30, 2007 [ADAMS Accession Number ML072610630].
7.35 NRC ANO Unit 2 Technical Specification Amendment 275, Control Element Assembly Drop Time, March 5, 2008 [ADAMS Accession Number ML080520223].
7.36 St. Lucie Unit 2 Letter L-2009-127, Technical Specification Modification Regarding Control Rod Assembly Drop Time, May 21, 2009 [ADAMS Accession Number ML091460050].
7.37 NRC St. Lucie Unit 2 Technical Specification Amendment 158, Control Element Assembly Drop Time, May 31, 2010 [ADAMS Accession Number ML100990275].
7.38 NE-002-020, CEA Insertion Time Measurement.
7.39 St. Lucie Unit 2 Letter L-2012-121, Information Regarding Fuel Thermal Conductivity Degradation Provided in Support of the Extended Power Uprate License Amendment Request, March 31, 2012 [ADAMS Accession Number ML12094A312].

Attachment 2 to

W3F1-2015-0040

Final Safety Analysis Report Revised Assessment

This attachment provides a detailed analysis of the revised Control Element Assembly (CEA) drop time impact on the Final Safety Analysis Report (FSAR) events. Waterford 3 utilizes the Westinghouse reload analysis methodology for each cycle.

FSAR 6.2 Containment Analyses

FSAR Section 6.2.1.3 describes the Loss of Coolant Accident (LOCA) mass and energy release analysis. FSAR Section 6.2.1.3 describes the Main Steam Line Break (MSLB) mass and energy release analysis. The LOCA and MSLB mass and energy releases are not impacted by the average CEA SCRAM insertion time curve change. The LOCA mass and energy release analysis shuts down on voiding in the core and the CEA SCRAM insertion time curve is not an input to this analysis. To confirm the expected impact of the increase of 0.2 seconds in average CEA SCRAM insertion time to 90% insertion increase on the MSLB mass and energy release analysis, the CEA insertion rate calculation was reviewed and confirmed to be negligibly affected.

FSAR 7.2.1.1.2.5 Core Protection Calculators and
FSAR 7.7.1.5 Core Operating Limit Supervisory System

Core Protection Calculator (CPC) is described in FSAR Section 7.2.1.1.2.5. Core Operating Limit Supervisory System (COLSS) is described in FSAR Section 7.7.1.5.

The COLSS and CPC functionality is used at Waterford-3 to measure plant parameters (e.g., temperature, pressure, flow, incore/excore detector signals) and utilize this information to make plant decisions. When outside of the limiting conditions for operation (LCO), the COLSS computer will provide alarms to the operator. When outside of the limiting safety system settings (LSSS), the CPC System (CPCS) will initiate a reactor trip.

COLSS and CPC are not impacted by the average or individual CEA SCRAM insertion time changes. Neither COLSS nor CPC provide any further action upon the onset of a reactor trip. Therefore, COLSS and CPC are not directly affected by a change in CEA SCRAM times and no evaluation was required.

The impact of the revised CEA SCRAM insertion curve on the FSAR Chapter 15 AORs supporting the Core Operating Limits Supervisory System (COLSS) and CPCS Non-LOCA transient analyses were reviewed with respect to the potential impact on the required overpower margin (ROPM), the minimum departure from nucleate boiling ratio (mDNBR), and the peak linear heat rate (LHR). The evaluation reviewed the transient events that support the CPCS filters and the neutron and thermal power penalties. Specifically, these analyses include: the Increased Main Steam Flow/Excess Load (EXLD) events, Control Element Assembly Withdrawal (CEAW) events, Single CEA Misoperation (single rod withdrawal) events, and the CPC filters analyses. The AORs were found to remain bounding.

The current Non-LOCA transient analyses neutron and thermal power penalty requirements for CPCS remain the same. The current Non-LOCA transient analyses ROPM requirements for COLSS remain the same.

FSAR 7.7.1.9 Reactor Power Cutback System

FSAR Section 7.7.1.9 describes the reactor power cutback system. The reactor power cutback event with the failure of the corresponding turbine runback is analyzed in support of COLSS and CPCS inputs. At the start of a turbine runback, selected lead banks are dropped into the core. If no turbine runback occurs by 18.5 seconds, the reactor is tripped. Upon reactor trip, a turbine trip occurs. The response time of the turbine trip and closure of the turbine stop valves is less than 0.3 second and the SCRAM rods begin to move 0.6 second after reactor trip. The time of mDNBR occurs at 0.1 second after the SCRAM rods begin to move. As the event is mitigated by the closure of the turbine stop valves, which terminates the primary-secondary power mismatch and mitigates the event, any negative reactivity supplements the mitigation of the event. Hence, this event is over as soon as negative reactivity is inserted into the core. Therefore there is no impact on the turbine runback event due to the implementation of the revised CEA SCRAM insertion curve.

FSAR 9.3.6 BTP 5-4 Natural Circulation Cooldown Analysis

FSAR Section 9.3.6.3.3 discusses the Branch Technical Position (BTP) 5-4 natural circulation cooldown analysis. Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the BTP 5-4 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. This means that RCS cooldown and associated consequences are not impacted by this change.

FSAR 9.5 Appendix R

FSAR Section 9.5 discusses the fire protection system. Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the fire protection evaluation. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. This means that RCS cooldown and associated consequences are not impacted by this change.

FSAR 15.1.1.1 Decrease in Feedwater Temperature

FSAR Sections 15.1.1.1 describes the decrease in feedwater temperature events. FSAR Sections 15.1.1.1 states that this event is comparable or less adverse (bounded) than the FSAR Sections FSAR Section 15.1.1.3 increased main steam flow event. The revised CEA drop time would not change the event characteristics so the increased main steam flow event remains the bounding event.

FSAR Section 15.1.1.1.5 describes the radiological consequences are less severe than those for the FSAR Section 15.1.2.4.5 inadvertent opening of a steam generator atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the inadvertent opening of a steam generator atmospheric dump valve event remains the bounding event.

FSAR 15.1.1.2 Increase in Feedwater Flow

FSAR Section 15.1.1.2 describes the increase in feedwater flow event. FSAR Sections 15.1.1.2 states that this event is no more adverse (bounded) than the FSAR Section 15.1.1.3 increased main steam flow event. The revised CEA drop time would not change the event characteristics so the increased main steam flow event remains the bounding event.

FSAR Section 15.1.1.2.5 describes the radiological consequences are less severe than those for the FSAR Section 15.1.2.4.5 inadvertent opening of a steam generator atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the inadvertent opening of a steam generator atmospheric dump valve event remains the bounding event.

FSAR 15.1.1.3 Increase Main Steam Flow

UFSAR Section 15.1.1.3 describes the increased main steam flow event. The increase in main steam flow event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation.

The time of peak power and the time that the CEAs begin to drop occur simultaneously. Thus, there is no impact on the peak linear heat rate due to the revised CEA drop time. The turbine admission valves are closing prior to the time that the SCRAM rods drop into the core. The reduction of steam flow is the dominating effect at reducing core power and mitigating the event. Hence impact of the revised CEA drop on the increased main steam flow event of minimum DNBR is insignificant.

The increased main steam flow event is a secondary cooldown event that results in a primary system cooldown. Thus, the primary and secondary pressures decrease from their initial value during the event and will remain below the (safety) design limits.

FSAR Section 15.1.1.3.5 describes the radiological consequences are less severe than those for the FSAR Section 15.1.2.4.5 inadvertent opening of a steam generator atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the inadvertent opening of a steam generator atmospheric dump valve event remains the bounding event.

FSAR 15.1.1.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

FSAR Section 15.1.1.4 describes the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve (IOADV) event. The IOADV event reactor trip occurs at 607 seconds due to low steam generator water level. FSAR Figure 15.1-19a shows that the minimum DNBR occurs prior to reactor trip, thus the revised CEA drop time would not impact minimum DNBR results. FSAR Figure 15.1-14 shows the maximum Reactor Coolant System (RCS) pressure and FSAR Figure 15.1-17 shows the maximum Steam Generator (SG) pressure. FSAR Figure 15.1-14 and Figure 15.1-17 show that primary and secondary peak pressures are limited by the safety valves and the revised CEA drop time would not impact the peak pressures. FSAR Section 15.1.1.4.5 states that the

radiological consequences are less severe than the consequences of the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Loss of Offsite Power event. The revised CEA drop time would not change the event characteristics so the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve with a Concurrent Loss of Offsite Power event remains the bounding radiological event.

FSAR 15.1.2.1 Decrease in Feedwater Temperature

FSAR Section 15.1.2.1 describes the decrease in feedwater temperature event. FSAR Section 15.1.2.1 states that this event is comparable or less adverse (bounded) than the FSAR Section 15.1.2.3 increased main steam flow event. The revised CEA drop time would not change the event characteristics so the increased main steam flow event remains the bounding event.

FSAR Section 15.1.2.1.5 describes the radiological consequences are less severe than those for the FSAR Section 15.1.2.4.5 inadvertent opening of a steam generator atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the inadvertent opening of a steam generator atmospheric dump valve event remains the bounding event.

FSAR 15.1.2.2 Increase in Feedwater Flow

FSAR Section 15.1.2.2 describes the increase in feedwater flow event. FSAR Section 15.1.2.2 states that this event is no more adverse (bounded) than the FSAR Section 15.1.2.3 increased main steam flow event. The revised CEA drop time would not change the event characteristics so the increased main steam flow events remain the bounding events.

FSAR Section 15.1.2.2.5 describes the radiological consequences are less severe than those for the FSAR Section 15.1.2.4.5 inadvertent opening of a steam generator atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the inadvertent opening of a steam generator atmospheric dump valve event remains the bounding event.

FSAR 15.1.2.3 Increased Main Steam Flow

FSAR Section 15.1.2.3 describes the increased main steam flow with a concurrent loss of offsite power event. The increase in main steam flow with a concurrent loss of offsite power event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

To partially offset the loss of DNBR margin due to the revised CEA drop time, analysis margin was reduced for the axial shape index (ASI) limits and the least negative Doppler reactivity. The analysis margin for the axial power distribution was reduced from an ASI of +0.3 to +0.2, which is conservative to the COLR limit of +0.16. The analysis margin for the least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{^{o}K}$. This still includes the UFSAR 15.0.3.3.1 uncertainty multiplier of 0.85 to conservatively

minimize the analysis Doppler impact. The reduction in least negative Doppler margin has an insignificant impact on the minimum DNBR.

The increase in peak secondary pressure is based on the loss of condenser vacuum results which showed an increase in peak secondary pressure of less than 1 psi.

The increase in main steam flow with a concurrent loss of offsite power event and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Secondary Pressure	1102 psia	1102 psia	1103 psia	<1210 psia
Minimum DNBR	1.071	1.071	1.051	NA
Calculated Fuel Failure	8%	< 4.0% (Cycle Specific)	< 5.0%	≤ 8%

The increase in main steam flow with a concurrent loss of offsite power event results demonstrate that the peak secondary pressure remains below the acceptance criteria. This event is not limiting with respect to peak linear heat rate. The calculated fuel failure remains below the radiological dose limit. The radiological doses remain within the acceptance criteria.

FSAR 15.1.2.3.5 describes the radiological consequences. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.1.2.3.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. This event assumes an 8% failed fuel limit and these results demonstrate that the fuel failure limits remain unchanged which means the radiological source terms remain the same. Thus, there is no change to the radiological results.

FSAR 15.1.2.4 Inadvertent Opening of a Steam Generator Atmospheric Dump Valve

FSAR Section 15.1.2.4 describes the inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component event. The inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

To offset the loss of DNBR margin due to the revised CEA drop time, analysis margin was reduced for the axial shape index (ASI) limits, the least negative Doppler reactivity, and the initial required overpower margin (ROPM). The analysis margin for the axial power distribution was reduced from an ASI of +0.3 to +0.2, which is conservative to the COLR limit of +0.16. The analysis margin for the least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{^{o}K}$. The reduction in least negative Doppler margin has an insignificant impact on the minimum DNBR. This still includes the FSAR 15.0.3.3.1 uncertainty multiplier of 0.85 to conservatively minimize the analysis Doppler impact. The initial analysis DNBR margin was reduced by increasing the analysis ROPM by 3% to the Core Operating Limits Supervisory System hot full power value.

The increase in peak primary pressure was based on the loss of condenser vacuum results which showed an increase in peak primary pressure of less than 1 psi when the pressurizer pressure exceeded the pressurizer safety valve opening setpoints (2575 psia). The increase in peak secondary pressure was based on the loss of condenser vacuum results which showed an increase in peak secondary pressure of less than 1 psi.

The inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component results and acceptance criteria are listed in the table below.

	FSAR	**Analysis of Record**	**Revised Analysis**	**Acceptance Criteria**
Peak Primary Pressure	2583 psia	2583 psia	2584 psia *	<2750 psia
Peak Secondary Pressure	1117 psia	1117 psia	1118 psia	<1210 psia
Minimum DNBR	≥1.24	1.247	1.277	≥ 1.24

The inadvertent opening of a steam generator atmospheric dump valve with a concurrent single failure of an active component event results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria. The minimum DNBR remains within the acceptance criteria. This event is not limiting with respect to peak linear heat rate. There are no changes required to the current COLSS database to support the revised CEA drop time.

FSAR 15.1.2.4.5 describes the radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.1.2.4.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. Thus, there is no change to the radiological results.

FSAR 15.1.3.1 Steam System Piping Failures Post-Trip Return-To-Power

UFSAR Section 15.2.3.1 describes the steam system piping failures post-trip return-to-power (R-t-P). The steam system piping failures post-trip R-t-P events were submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluations started with the analysis of record and only revised the CEA drop time to determine the impact.

The loss of feedwater flow event demonstrated that the impact of the revised CEA drop time on long term parameters is insignificant. Hence there is an insignificant impact on the plant cooldown to shutdown cooling conditions post-trip.

The steam system piping failures post-trip R-t-P hot full power (HFP) and hot zero power (HZP) with and without loss of offsite power (LOOP) for inside containment (IC) and outside containment (OC) events are dominated by the total scram reactivity added and the rate of the primary side cooldown and its associated moderator and fuel feedback effects. The total reactivity added has not changed due to the revised CEA drop time.

The primary side cooldown is directly dependent on the secondary side cooldown. The secondary side cooldown is dependent on the replacement steam generator integral flow restrictors, which make the IC and OC breaks the same event. Since the steam generator blowdown is not impacted by the revised CEA drop time, there is no change in the secondary side cooldown. As there is no change to the secondary side cooldown, there is no change in the primary side cooldown. Hence, there is an insignificant, if any, impact on the steam system piping failures events.

For the steam system piping failures post-trip R-t-P HFP and HZP with LOOP for the IC and OC events, the results and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Hot Full Power				
Maximum post-trip fission power	2.1% of 3716 MWt	2.1% of 3716 MWt	2.1% of 3716 MWt	NA
Maximum post-trip reactivity	0.0026 %Δρ	0.0026 %Δρ	0.0026 %Δρ	NA
Minimum DNBR (MacBeth)	1.47	1.47	1.47	≥1.30 for OC only
Hot Zero Power				
Maximum post-trip fission power	2.5% of 3716 MWt	2.5% of 3716 MWt	2.5% of 3716 MWt	NA
Maximum post-trip reactivity	0.1324 %Δρ	0.1324 %Δρ	0.1324 %Δρ	NA
Minimum DNBR (MacBeth)	1.61	1.61	1.61	≥1.30 for OC only
Fuel Failure **Inside Containment** **Outside Containment**	 2% 0%	 0% 0%	 0% 0%	 ≤ 2% 0%

For steam system piping failures post-trip R-t-P HFP and HZP with no LOOP for IC and OC events, the results and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Hot Full Power				
Maximum post-trip fission power	3.45% of 3716 MWt	3.45% of 3716 MWt	3.45% of 3716 MWt	NA
Maximum post-trip reactivity	-0.048 %Δρ	-0.048 %Δρ	-0.048 %Δρ	NA
Maximum post-trip linear heat generation rate (LHGR)	18.4 kW/ft	18.4 kW/ft	18.4 kW/ft	≤ 24 kW/ft
Hot Zero Power				
Maximum post-trip fission power	4.9% of 3716 MWt	4.9% of 3716 MWt	4.9% of 3716 MWt	NA
Maximum post-trip reactivity	0.1641 %Δρ	0.1641 %Δρ	0.1641 %Δρ	NA
Maximum post-trip LHGR	21.25 kW/ft	21.25 kW/ft	21.25 kW/ft	≤ 24 kW/ft
Fuel Failure	0%	0%	0%	0%

The steam system piping failures post-trip R-t-P results, in an insignificant impact on the maximum post-trip fission power, maximum post-trip reactivity, post-trip minimum DNBR, and maximum post-trip linear heat rate.

FSAR Section 15.1.3.1.5 describes the steam line break post trip return to power radiological consequences. The radiological consequences of the outside containment

Post-Trip Main Steam Line Break are less severe than the results of a FSAR Section 15.2.3.1.3.1.5 Feedwater System Pipe Break. The radiological consequences of the inside containment Post-Trip Main Steam Line Break are analyzed in conjunction with the FSAR Section 15.1.3.3.5 Pre-Trip Main Steam Line Break.

FSAR 15.1.3.2 Steam System Piping Failures Inside and Outside Containment Modes 3 and 4 with All CEAs on the Bottom

UFSAR Section 15.1.3.2 describes the steam system piping failures inside and outside containment Modes 3 and 4 with all CEAs on the bottom. Since all CEAs have been already inserted, the revised CEA drop time would have no impact on the MSLB event inside and outside containment for Modes 3 and 4. Thus, there is no impact UFSAR Section 15.1.3.2 results and conclusions.

FSAR 15.1.3.3 Steam System Piping Failures: Pre-Trip Power Excursion Analysis

UFSAR Section 15.1.3.3 describes the steam system piping failures: pre-trip power excursion. The steam system piping failures: pre-trip power excursion events were submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluations started with the analysis of record and only revised CEA drop time to determine the impact.

The loss of DNBR margin due to the revised CEA drop time is applied against the DNBR analysis margin between the analysis of record DNBR and the DNB SAFDL value of 1.24. The revised CEA drop time resulted in less than a 33% loss of DNBR margin.

The peak core power occurs prior to when the CEAs begin to drop for both the IC and OC pre-trip steamline break events, so there is no impact on maximum peak linear heat. Thus, the maximum peak linear heat rate remains within the acceptance criteria.

The steam system piping failures: pre-trip power excursion events are a secondary side cooldown event that results in a primary system cooldown. Thus, the primary and secondary pressures decrease from their initial value during the event and will remain below the (safety) design limits.

The loss of feedwater flow event demonstrated that the impact of the revised CEA drop time on long term parameters is insignificant. Hence there is an insignificant impact on the plant cooldown to shutdown cooling conditions post-trip.

The steam system piping failures: pre-trip power excursion IC event and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Minimum DNBR	1.428	1.428	1.365	NA
Calculated Fuel Failure	8%(1) (2)	0%	0%	≤ 8% (2)

(1) 8% is used for the radiological doses, but zero was calculated.
(2) Radiological doses are based upon 10%; 2% from the post-trip steamline IC break in Section 15.1.3.1 and 8% from the pre-trip excursion IC break in Section 15.1.3.3.

The steam system piping failures: pre-trip power excursion OC event and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Minimum DNBR	1.451	1.451	1.381	≥ 1.24
Calculated Fuel Failure	0%	0%	0%	0%

The calculated fuel failure remains below the radiological dose limit.

FSAR Section 15.1.3.3.5 describes the steam line break pre-trip event radiological consequences. The radiological consequences of the outside containment Pre-Trip Main Steam Line Break are bounded by the results reported for a Feedwater System Pipe Break (See Subsection 15.2.3.1.3.1.5). This event assumes a 10% failed fuel limit (8% fuel failure due to the pre-trip SLB and 2% fuel failure due to the post trip SLB) and the revised CEA drop time does not change the failed fuel consequences.

The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.1.3.3.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. This event assumes a 10% failed fuel limit and these results demonstrate that the fuel failure limits remain unchanged which means the radiological source terms remain the same. Thus, there is no change to the radiological results.

FSAR 15.2.1.1 Loss of External Load

FSAR Sections 15.2.1.1 describes the loss of external load event. FSAR Sections 15.2.1.1 states that these events are no more adverse (bounded) than the FSAR Sections 15.2.1.3 loss of condenser vacuum event. FSAR Sections 15.2.1.1 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so loss of external load event remains bounded by existing FSAR events.

FSAR 15.2.1.2 Turbine Trip

FSAR Section 15.2.1.2 describes the turbine trip event. FSAR Section 15.2.1.2 states that this event is no more adverse (bounded) than the FSAR Sections 15.2.1.3 loss of condenser vacuum event. FSAR Section 15.2.1.2 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the turbine trip event remains bounded by existing FSAR events.

FSAR 15.2.1.3 Loss of Condenser Vacuum

UFSAR Sections 15.2.1.3 and 15.2.2.3 describe the LOCV event. The LOCV event was submitted and approved for the extended power uprate (Reference B-4). Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation (Reference B-5). The revised evaluation started with the AOR and only revised CEA drop time to determine the impact.

The loss of condenser vacuum results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	2711 psia	2711 psia	2712 psia	<2750 psia
Peak Secondary Pressure	1181 psia	1180 psia	1180 psia	<1210 psia
Minimum DNBR	>1.24	>1.24	>1.24	≥ 1.24
Maximum PLHR	<21 kw/ft	<21 kw/ft	<21 kw/ft	≤ 21 kW/ft
Time to Pressurizer Fill	>15 minutes	1029.9 seconds	990.2 seconds	>900 seconds
Peak Secondary Pressure for 1 Failed MSSV	NA	1186 psia	1186 psia	<1210 psia
Peak Secondary Pressure for 2 Failed MSSVs	NA	1190 psia	1188 psia	<1210 psia

The loss of condenser vacuum event results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only a minimal increase in peak primary pressure. The minimum DNBR and maximum peak linear heat reate (PLHR) remain within the acceptance criteria. For the long term pressurizer fill case, there still exists sufficient time (minimum of 15 minutes after the reactor trip) for operator action to mitigate the event prior to water covering the primary safety valve inlet nozzles. For the one and two inoperable main steam safety valves per steam generator cases, the peak secondary pressures remained below the secondary pressure acceptance criteria.

FSAR 15.2.1.4 Loss of Normal AC Power

FSAR Section 15.2.1.4 describes the loss of normal AC power event. FSAR Sections 15.2.1.4 states that the fuel performance aspects of this event are bounded by the FSAR

Section 15.3.2.1total loss of forced reactor coolant flow event. The peak pressure aspects of this event are bounded by the FSAR Section 15.2.1.3 loss of condenser vacuum event. FSAR Section 15.2.1.4 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the loss of normal AC power events remain bounded by existing FSAR events.

FSAR 15.2.1.5 Steam Pressure Regulator Failure

FSAR Section 15.2.1.5 (Reference 1) describes the steam pressure regulator failure event. FSAR 15.2.1.5 states that this transient is less severe than the FSAR 15.2.1.3 loss of condenser vacuum event. The revised CEA drop time would not change the event characteristics so the steam pressure regulator failure event remains bounded by existing FSAR event.

FSAR 15.2.2.1 Loss of External Load

FSAR Sections 15.2.2.1 describes the loss of external load event. FSAR Section 15.2.2.1 states that this event is no more adverse (bounded) than the FSAR Section 15.2.2.3 loss of condenser vacuum event. FSAR Section 15.2.2.1 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so loss of external load events remain bounded by existing FSAR events.

FSAR 15.2.2.2 Turbine Trip

FSAR Section 15.2.2.2 describes the turbine trip event. FSAR Section 15.2.2.2 states that this event is no more adverse (bounded) than the FSAR Sections 15.2.2.3 loss of condenser vacuum event. FSAR Section 15.2.2.2 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the turbine trip events remain bounded by existing FSAR events.

FSAR 15.2.2.3 Loss of Condenser Vacuum

FSAR 15.2.2.3 states that this event is bounded by the FSAR Section 15.2.1.3 Loss of Condenser Vacuum.

FSAR 15.2.2.4 Loss of Normal AC Power

FSAR Section 15.2.2.4 describes the loss of normal AC power event. FSAR Sections 15.2.2.4 states that the fuel performance aspects of these events are bounded by the FSAR Section 15.3.2.1 total loss of forced reactor coolant flow event. The peak pressure aspects of this event is bounded by the FSAR Section 15.2.1.3 loss of condenser vacuum event. The revised CEA drop time would not change the event characteristics so the loss of normal AC power events remain bounded by existing FSAR events.

FSAR 15.2.2.5 Loss of Normal Feedwater Flow

UFSAR Section 15.2.2.5 describes the loss of normal feedwater flow event. The loss of normal feedwater event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the AOR and only revised CEA drop time to determine the impact.

The loss of feedwater flow results and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	2248 psia	2248 psia	2268 psia	<2750 psia
Peak Secondary Pressure	1170 psia	1071 psia	1072 psia	<1210 psia
Minimum DNBR	>1.24	>1.24	>1.24	≥ 1.24
Maximum Peak LHR	<21 kW/ft	<21 kW/ft	<21 kW/ft	≤ 21 kW/ft
Minimum SG Water Inventory	43,030 lbm	43,030 lbm	42,830 lbm	NA

The loss of feedwater results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only small increases in the peak pressure values. The minimum DNBR and maximum peak LHR remain within the acceptance criteria.

FSAR 15.2.3.1 Feedwater System Pipe Breaks

UFSAR Section 15.2.3.1 describes the feedwater system pipe break events. The feedwater system pipe break events were submitted and approved for the extended power uprate (Reference B-1). Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the new departure from nucleate boiling ratio (DNB, DNBR) correlation (Reference B-2). The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

For large feedwater system pipe break events, the results and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Peak Primary Pressure	2745 psia	2744.6 psia	2745.7 psia	<3000 psia
Peak Secondary Pressure	1122.1 psia	1122.1 psia	1123.1 psia	<1210 psia
Effective EFW Flow to Unaffected SG	575 gpm	575 gpm	575 gpm	No dry-out of Unaffected SG

For small feedwater system pipe break events, the results and acceptance criteria are listed in the table below.

	UFSAR	Analysis of Record	Revised Evaluation	Acceptance Criteria
Peak Primary Pressure	2655.4 psia	2655.4 psia	2656.5 psia	<2750 psia
Peak Secondary Pressure	1119.5 psia	1119.5 psia	1120.5 psia	<1210 psia
Effective EFW Flow to Unaffected SG	575 gpm	575 gpm	575 gpm	No dry-out of Unaffected SG

The feedwater system pipe break results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only a minimal increase in peak primary pressure. The minimum DNBR and maximum peak linear heat rate (PLHR) remain within the acceptance criteria. The 575 gpm of EFW remains sufficient to prevent dryout of the unaffected steam generator.

FSAR Section 15.2.3.1.3.1.5 describes the feedwater line break radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.2.3.1.3.1.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. Thus, there is no change to the radiological results.

FSAR 15.2.3.2 Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System

UFSAR Section 15.2.3.2 describes the loss of normal feedwater flow event. The loss of normal feedwater event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the AOR and only revised CEA drop time to determine the impact.

The loss of feedwater flow plus single failure of the steam bypass control system results and acceptance criteria are listed in the following table.

	UFSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	NA	2171 psia	2191 psia	<2750 psia
Peak Secondary Pressure	NA	1053 psia	1054 psia	<1210 psia
Minimum DNBR	>1.24	>1.24	>1.24	≥ 1.24
Maximum Peak LHR	<21 kW/ft	<21 kW/ft	<21 kW/ft	≤ 21 kW/ft
Minimum SG Water Inventory	9313 lbm	9313 lbm	9113 lbm	NA

The loss of feedwater flow plus single failure of the pressurizer level control system results at 30 minutes (approximately 29 minutes after reactor trip) are listed in the following table.

	Units	UFSAR	Analysis of Record	Revised Analysis
Core Power	% of rated power	NA	1.687	1.687
RCS Average Temperature	ºF	NA	558.4	557.8
Pressurizer Pressure	psia	NA	2056	2040
Pressurizer Liquid Mass	lbm	NA	52,750	52,468
SG Pressure	psia	NA	1104	1099
SG Liquid Mass	lbm	NA	62,260 (SG 1) 62,270 (SG 2)	62,080 (SG 1) 62,090 (SG 2)
Integrated Emergency Feedwater Flow	lbm	NA	56,460 (SG 1) 55,390 (SG 2)	56,950 (SG 1) 55,710 (SG 2)
Total Steam Released from Turbine and MSSVs	lbm	NA	369,000	370,600

The loss of feedwater results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only small increases in the peak pressure values. The minimum DNBR and maximum peak LHR remain within the acceptance criteria. For the long-term cooling event, the differences at 30 minutes (29 minutes after reactor trip) are insignificant such that the subsequent transient is the same.

FSAR 15.3.1.1 Partial Loss of Forced Reactor Coolant Flow

FSAR Section 15.3.1.1 describes the partial loss of forced reactor coolant flow event. FSAR Section 15.3.1.1 states that consequences of a partial loss of forced reactor

coolant flow are no more adverse than those following the FSAR Section 15.3.2.1 total loss of forced reactor coolant flow. FSAR Section 15.3.1.1 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the partial loss of forced reactor coolant flow event remains bounded by existing FSAR events.

FSAR 15.3.2.1 Total Loss of Forced Reactor Coolant Flow

FSAR Section 15.3.2.1 describes the total loss of forced reactor coolant event. The total loss of forced reactor coolant event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

To partially offset the loss of DNBR margin due to the revised CEA drop time, analysis margin was reduced for the axial shape index (ASI) limits and the least negative Doppler reactivity. The analysis margin for the axial power distribution was reduced from an ASI of +0.3 to +0.2, which is conservative to the COLR limit of +0.16. The analysis margin for the least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{^{o}K}$. This still includes the UFSAR 15.0.3.3.1 uncertainty multiplier of 0.85 to conservatively minimize the analysis Doppler impact. The reduction in least negative Doppler margin has an insignificant impact on the minimum DNBR.

The increase in peak primary pressure was based on the loss of feedwater long term results which showed an increase in peak primary pressure of less than 20 psi when the pressurizer pressure remained below the pressurizer safety valve opening setpoints (2575 psia).

The total loss of forced reactor coolant results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	2395 psia	2395 psia	2415 psia	<2750 psia
Minimum DNBR	>1.24	1.302	1.293	≥ 1.24

The total loss of forced reactor coolant event results demonstrate that the peak primary pressure remains below the acceptance criteria with only a small increase in peak primary pressure. The minimum DNBR remains within the acceptance criteria. This event is not limiting with respect to peak linear heat rate. There are no changes required to the current COLSS database to support the revised CEA drop time.

FSAR 15.3.2.2 Partial Loss of Forced Reactor Coolant Flow

FSAR Section 15.3.2.2 describes the partial loss of forced reactor coolant flow event. FSAR Section 15.3.2.2 states that consequences of a partial loss of forced reactor coolant flow are no more adverse than those following the FSAR Section 15.3.2.1 total loss of forced reactor coolant flow. FSAR Section 15.3.2.2 states that the radiological consequences due to steam releases from the secondary system are less severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of the atmospheric dump valve. The revised CEA drop time would not change the event characteristics so the partial loss of forced reactor coolant flow event remains bounded by existing FSAR events.

FSAR 15.3.3.1 Single Reactor Coolant Pump (RCP) Shaft Seizure / Sheared Shaft

FSAR Section 15.3.3.1 describes the single reactor coolant pump (RCP) shaft seizure / sheared shaft events. The single RCP shaft seizure / sheared shaft events were submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

To partially offset the loss of DNBR margin due to the revised CEA drop time, analysis margin was reduced for the axial shape index (ASI) limits and the least negative Doppler reactivity. The analysis margin for the axial power distribution was reduced from an ASI of +0.3 to +0.2, which is conservative to the COLR limit of +0.16. The analysis margin for the least negative Doppler reactivity was reduced from -0.00113 to -0.0013 $\Delta\rho/\sqrt{^{o}K}$. This still includes the UFSAR 15.0.3.3.1 uncertainty multiplier of 0.85 to conservatively minimize the analysis Doppler impact. The reduction in least negative Doppler margin has an insignificant impact on the minimum DNBR.

The increase in peak primary pressure was based on the loss of feedwater long term results which showed an increase in peak primary pressure of less than 20 psi when the pressurizer pressure remained below the pressurizer safety valve opening setpoints (2575 psia). The increase in peak secondary pressure was based on the loss of condenser vacuum results which showed an increase in peak secondary pressure of less than 1 psi.

The single RCP shaft seizure / sheared shaft results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	2422 psia	2422 psia	2442 psia	<2750 psia
Peak Secondary Pressure	1117 psia	1117 psia	1118 psia	<1210 psia
Minimum DNBR	1.1450	1.1450	1.131	NA
Calculated Fuel Failure	15%	< 8.0% (Cycle Specific)	< 10.5%	≤ 15%

The single RCP shaft seizure / sheared shaft events results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only a small increase in peak primary and secondary pressure. This event is not limiting with respect to peak linear heat rate. The calculated fuel failure remains below the radiological dose limit.

FSAR section 15.3.3.1.4.1 describes the radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.3.3.1.4.1 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release remain the same. This event assumes a 15% failed fuel limit and these results demonstrate that the fuel failure limits remain unchanged which means the radiological source terms remain the same. Thus, there is no change to the radiological results.

FSAR 15.4.1.1 Uncontrolled CEA Withdrawal from a Subcritical

UFSAR Section 15.4.1.1 describes the uncontrolled CEA withdrawal from subcritical conditions event. The uncontrolled CEA withdrawal from subcritical conditions event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

The analysis of record for the UFSAR uncontrolled CEA withdrawal from subcritical conditions event calculated a minimum DNBR for the NGF DNBR correlation. As part of a parametric evaluation the analysis of record increased the full power heat flux by 13% and reduced the core average mass flux by 5%. The combination of increase heat flux and reduced mass flux more than offsets the very small increase in heat flux due to the revised CEA drop time.

For the uncontrolled CEA withdrawal from subcritical conditions event, the full-power seconds limit was calculated based on a maximum fuel temperature of 3500 ºF. This fuel temperature limit is well under the bounding fuel centerline temperature limit of 4663ºF.

The loss of condenser vacuum event peak primary and secondary pressure increases would bound that expected for the uncontrolled CEA withdrawal from subcritical conditions. Thus the primary and secondary pressures will remain well below the (safety) design limits.

The uncontrolled CEA withdrawal from subcritical conditions results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Minimum DNBR	≥ 1.24	4.31	3.66	≥ 1.24
Full-Power Seconds Limit	NA	0.44 second	0.56 second	≤ 1.59 seconds
Fuel Centerline Temperature	< 4900 ºF[Note 1]	< 3500 ºF	< 3500 ºF	< 4663 ºF

The minimum DNBR and maximum peak fuel centerline temperature remain within the acceptance criteria.

Note 1: That the UFSAR Section 15.4.1.1.3 value of 4900 ºF is incorrect and will be revised.

FSAR 15.4.1.2 Uncontrolled CEA Withdrawal at Low Power

UFSAR Section 15.4.1.2 describes the uncontrolled CEA withdrawal from low power conditions event. The uncontrolled CEA withdrawal from low power conditions event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

The analysis of record for the UFSAR uncontrolled CEA withdrawal from low power conditions event calculated a minimum DNBR value for the NGF DNBR correlation. As part of a parametric evaluation the analysis of record increased the full power heat flux by 13% and reduced the core average mass flux by 5%. The combination of increase heat flux and reduced mass flux more than offsets the very small increase in heat flux due to the revised CEA drop time.

For the uncontrolled CEA withdrawal from low power conditions event, the full-power seconds limit was calculated based on a maximum fuel temperature of 3500 ºF. This fuel temperature limit is well under the bounding fuel centerline temperature limit of 4663 ºF.

The loss of condenser vacuum event peak primary and secondary pressure increases would bound that expected for the uncontrolled CEA withdrawal from low power condition. Thus the primary and secondary pressures will remain well below the (safety) design limits.

The uncontrolled CEA withdrawal from low power conditions results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Minimum DNBR	≥ 1.24	3.44	2.90	≥ 1.24
Full-Power Seconds Limit	NA	1.35 second	1.49 second	≤ 2.53 seconds
Fuel Centerline Temperature	< 4900 ºF[Note 1]	< 3500 ºF	< 3500 ºF	< 4663 ºF

The minimum DNBR and maximum peak fuel centerline temperature remain within the acceptance criteria.

Note 1: That the UFSAR Section 15.4.1.1.3 value of 4900 ºF is incorrect and will be revised.

FSAR 15.4.1.3 Uncontrolled CEA Withdrawal at Power

UFSAR Section 15.4.1.3 describes the uncontrolled CEA withdrawal at power event. The uncontrolled CEA withdrawal at power event was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation.

The time of minimum DNBR and the time of peak linear heat rate occur prior to the time the CEAs begin to drop into the core. Hence there is no impact of the revised CEA drop on the CEA withdrawal event.

The loss of condenser vacuum event peak primary and secondary pressure increases would bound that expected for the uncontrolled CEA withdrawal from at power conditions. Thus the primary and secondary pressures will remain well below the (safety) design limits.

FSAR 15.4.1.4 CEA Misoperation

UFSAR Section 15.4.1.4 describes the CEA misoperation events. These events include the CEA single and subgroup drop events and the single CEA withdrawal (SCEAW) from within the CPCS deadband event. The single CEA and subgroup drop events do not generate a reactor trip, thus the revised CEA drop time would not impact these events. The Single CEA withdrawal event is bounded by the current AOR, thus the revised CEA drop time would not impact these events.

FSAR 15.4.1.5 CVCS Malfunction (Inadvertent Boron Dilution)

FSAR Section 15.4.1.5 describes the chemical and volume control system malfunction (CVCS) event or inadvertent boron dilution event. This event is analyzed for all modes of plant operation. The operational mode 1 and 2 inadvertent boron dilution event is bounded by the FSAR Section 15.4.1.2 and 15.4.1.3 hot full power (HFP) and hot zero power (HZP) CEA Bank Withdrawal (CEAW) events. The revised CEA drop time would not change the event characteristics so the inadvertent boron dilution event remains bounded by existing FSAR events.

For the inadvertent boron dilution event in operational Modes 3, 4, 5, and 6, all CEAs have been already inserted. Thus, the revised CEA drop time would have no impact on the inadvertent boron dilution event.

FSAR 15.4.1.6 Startup of an Inactive Reactor coolant System Pump

FSAR Section 15.4.1.6 describes the start of an inactive reactor coolant system pump in operational modes 3, 4 or 5 with all CEAs on the bottom. Technical Specification 3.4.1.1

requires all reactor coolant pumps to be operating in modes 1 and 2. In operational mode 6, the RCPs breakers are required to be open. Since all CEAs are inserted in Operational Modes 3, 4, and 5, there is no impact due to the revised CEA drop time.

FSAR 15.4.1.7 Uncontrolled CEA Withdrawal from a Subcritical Condition – Mode 3, 4, 5 With All CEAS on the Bottom

FSAR Section 15.4.1.7 describes the uncontrolled CEA withdrawal from a subcritical condition - operating modes 3, 4, and 5 with all CEAs on the bottom. The withdrawal of a CEA adds reactivity to the reactor core causing the core power level to increase, as well as a time dependent redistribution of core power. The CPCS generates a reactor power trip when the CPCS bypass is automatically removed at 10^{-4}% of rated thermal power. This trip causes the shutdown of the reactor prior to the time that sensible heat is generated in the core. The CPCS bypass power is not impacted by the revised CEA drop time curve. The FSAR discussion remains valid.

FSAR 15.4.3.1 Inadvertent Loading of a Fuel Assembly into the Improper Position

FSAR Section 15.4.3.1 describes the inadvertent loading of fuel assembly into the improper position event. The inadvertent loading of fuel assembly into the improper position could cause a difference in the local power peaking for the misloaded core. This event is analyzed with respect to thermal margin for power operations. Since, the event does not assume a reactor trip, the revised CEA drop time would have no impact.

FSAR 15.4.3.2 CEA Ejection

FSAR Section 15.4.3.2 describes the control element assembly (CEA) ejection events. The CEA ejection events were submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation. The revised evaluation started with the analysis of record and only revised CEA drop time to determine the impact.

The CEA ejection event results and acceptance criteria are listed in the table below.

	FSAR	Analysis of Record	Revised Analysis	Acceptance Criteria
Peak Primary Pressure	2631 psia	2631 psia	2632 psia	<2750 psia
Fuel rods that experience DNB clad damage	≤ 15%	≤ 11.6%	≤ 12.1%	≤ 15%
Fuel rods having radial average fuel enthalpy ≥ 280 cal/gm	0	0	0	0
Fuel rods having centerline enthalpy ≥ 231.3 cal/gm	0	0	0	0

The CEA ejection event results demonstrate that the peak primary pressure remains below the acceptance criterion with only a small increase in peak primary pressure. The

maximum fuel rod radial average and maximum incipient centerline melting enthalpy remain within the acceptance criteria. The total calculated fuel failures remain below the radiological dose limits.

FSAR 15.4.3.2.5 describes the CEA Ejection radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.4.3.2.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam releases remain the same. This event assumes a 15% failed fuel limit and these results demonstrate that the fuel failure limits remain unchanged which means the radiological source terms remain the same. Thus, there is no change to the radiological results.

FSAR 15.5.1.1 CVCS Malfunction

FSAR Section 15.5.1.1 describes the chemical and volume control system (CVCS) malfunction event. FSAR Section 15.5.1.1.3.b states that since the pressure transient is due to an increase in primary coolant inventory and not to thermal expansion, no power, coolant temperature, or DNB transient is produced prior to reactor trip. Therefore, the initial conditions for the principal process variables monitored by the COLSS, with the exception of RCS pressure, have no effect on the consequences of this transient. This means that the CEA insertion timing is not critical provided the reactor trip occurs and the negative reactivity is inserted. Thus, the revised CEA drop time has no impact on these analyses.

FSAR 15.5.1.2 Inadvertent Operation of the ECCS During Power Operation

FSAR Section 15.5.1.2 describes the inadvertent operation of the emergency core cooling system (ECCS) during power operation event. FSAR Section 15.5.1.2 specifically states that during power operation, the CPC low pressure trip ensures that the RCS pressure is greater than 1785 psia. This pressure exceeds the shutoff head of the safety injection pumps and the opening pressure of the safety injection tanks. Therefore, a spurious safety injection actuation signal (SIAS) will not cause injection of emergency cooling fluid into the RCS during power operation. In addition, Technical Specification 3.2.8 limiting conditions for operation requires a minimum primary pressure of 2125 psia in operational modes 1 and 2. This means this event is not credible in modes 1 and 2, therefore the revised CEA drop time has no impact on this event.

FSAR 15.5.2.1 CVCS Malfunction

FSAR Section 15.5.2.1 describes the chemical and volume control system (CVCS) malfunction event. FSAR Section 15.5.1.1.3.b states that since the pressure transient is due to an increase in primary coolant inventory and not to thermal expansion, no power, coolant temperature, or DNB transient is produced prior to reactor trip. Therefore, the initial conditions for the principal process variables monitored by the COLSS, with the exception of RCS pressure, have no effect on the consequences of this transient. This

means that the CEA insertion timing is not critical provided the reactor trip occurs and the negative reactivity is inserted. Thus, the revised CEA drop time has no impact on these analyses.

FSAR 15.6.3.1 Primary Sample or Instrument Line Break

UFSAR Section 15.6.3.1 describes the primary sample or instrumentation line pipe break outside of containment. The letdown line break event is analyzed at Hot Full Power (HFP) to determine the amount of primary mass released from a break in the letdown line pipe and ensures it bounds all other sample and instrument line breaks outside of containment. This event is performed as a parametric in break area such that a reactor trip occurs simultaneous with operator action at 1800 seconds. At this time, the operator terminates the primary mass release by isolating the letdown line and trips the reactor if an automatic trip has not occurred. The key transient result is the total primary mass released due to the letdown line pipe break prior to isolation and is required to ensure that the radiological dose criteria are met. The radiological doses are strongly dependent on the primary mass release and to a smaller degree on secondary releases. Independent of an automatic or operator initiated reactor trip, the primary operator action is the isolation of the letdown line and termination of the primary mass at 1800 seconds. Since the letdown isolation occurs simultaneously with reactor trip, the letdown line isolation occurs prior to the SCRAM rods dropping. The primary side mass release from the letdown line which is the largest contributor to the radiological doses is not impacted by the revised CEA SCRAM insertion curve.

The secondary side contribution to the radiological doses is not impacted as the Loss of Feedwater Flow (LOFW) event demonstrated that there is no impact on the secondary plant responses at 30 minutes. Radiological doses are calculated over a two and eight hour periods, and / or until shutdown cooling conditions are reached. As all of these times are much longer than the 30 minute duration of the LOFW long term cooldown transient, the probability that the final plant responses are impacted is zero. The LOFW also demonstrated that the power fraction 29 minutes after reactor trip was the same for the current and revised CEA SCRAM insertion curves. Hence, the impact from the post-trip decay heat is the same. At 30 minutes, the LOFW event demonstrated that post-trip parameters were the same; primary temperature (< 1 ºF difference), the SG liquid inventory (<0.3% difference), steam generator pressure (< 6 psi), main steam safety valve steam releases (<1%), and emergency feedwater mass added (< 1%). Thus, the energy stored in the primary metal mass, RCS liquid mass, and SG liquid mass is the same. Hence the amount of energy needed to be removed to reach shutdown cooling is not changed, thus input to the radiological doses is the same. There is no impact due to the revised CEA SCRAM insertion curve.

FSAR 15.6.3.2 Steam Generator Tube Rupture

UFSAR Section 15.6.3.2 describes the steam generator tube rupture events. The steam generator tube rupture events were submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation.

FSAR Section 15.6.3.2.1.4 describes the steam generator tube rupture radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.6.3.2.1.4 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. Thus, there is no change to the radiological results.

FSAR 15.6.3.3 Loss of Coolant Accident

FSAR Section 15.6.3.3 describes the Loss of Coolant Accidents (LOCA). The Large Break LOCA and Post-LOCA Long Term Cooling Analyses do not explicitly model or credit core reactivity due to CEA insertion, thus the revised CEA drop time has no impact.

The Small Break LOCA analysis does credit CEA rod insertion. The Small Break LOCA analysis of record (AOR) CEA reactivity curve was compared to the revised CEA drop time curve. The current Small Break LOCA analysis reactivity curve bounds that of the revised CEA drop time curve, thus there is no impact to the AOR.

FSAR Section 15.6.3.3.5 describes the loss of coolant accident radiological consequences. The radiological consequences were not adversely impacted by this change. The Loss of Normal Feedwater Flow event (FSAR 15.2.3.2) was chosen to evaluate the transient characteristics with respect to energy deposition and associated steam releases which would be applicable to the FSAR 15.6.3.3.5 analysis. The analysis showed that the differences in primary and secondary system energy after reactor trip is insignificant. As time increases farther past the time of CEA rod insertion, the differences of the impact of the revised CEA drop time become zero. The radiological releases due to steam release and break flow would remain the same. Thus, there is no change to the radiological results.

FSAR 15.6.3.4 Inadvertent Opening of a Pressurizer Safety Valve

FSAR Section 15.6.3.4 describes that the results of an inadvertent opening of a pressurizer safety valve event are bounded by the results of the Small Break LOCA AOR in the reactor coolant pump discharge leg. The revised CEA drop time has no impact on the Small Break LOCA AOR. Therefore, the results for the limiting Small Break LOCA continue to bound the results of an inadvertent opening of a pressurizer safety valve event.

FSAR 15.7.3.3 Postulated Radioactive Releases Due to Liquid Containing Tank Failures

FSAR Section 15.7.3.3 describes the postulated radioactive releases due to liquid containing tank failures. Since the postulated tank failure does not required CEA insertion to mitigate the event, the revised CEA drop time will have no impact on this event.

FSAR 15.7.3.4 Design Basis Fuel Handling Accidents

FSAR Section 15.7.3.4 describes the fuel handling accident. The fuel handling accident analysis evaluates the case of a fuel bundle being dropped from the fuel handling device and impacting one fuel bundle located in the spent fuel rack or one or more fuel bundles located in the reactor core. CEA insertion is not required for event mitigation, therefore the revised CEA drop time has no impact on this event.

FSAR 15.7.3.5 Spent Fuel Cask Drop Accidents

FSAR Section 15.7.3.5 describes the spent fuel cask drop accidents. FSAR Section 9.1.4 discusses that the cask handling crane and other heavy loads. CEA insertion is not required for event mitigation, therefore the revised CEA drop time has no impact on this event.

FSAR 15.8 Anticipated Transients Without Scram

UFSAR Section 15.8 describes the anticipated transients without scram (ATWS) events. The limiting ATWS event is the limiting peak RCS pressure anticipated operational occurrence. The Loss of Load (LOL) event is analyzed to demonstrate that the diverse scram system (DSS) generates a reactor trip that in conjunction with pressurizer safety valves (PSVs) mitigates the consequences of the LOL event. The LOL event is the same as the Loss of Condenser Vacuum (LOCV) event, but without the simultaneous loss of main feedwater on turbine trip. The peak primary pressure post-trip is controlled by the opening of the pressurizer safety valves (PSVs). This is confirmed by the LOCV event, which was analyzed to justify the peak primary and secondary side pressure responses. The rate of negative reactivity insertion during the SCRAM insertion after reactor trip is unimportant, since the primary function of the DSS is to trip the reactor and insert the SCRAM rods. Implementation of the revised CEA SCRAM insertion curve has no impact on the DSS setpoints and no impact on the insertion of the SCRAM rods. Hence there is no impact due to the revised CEA SCRAM insertion curve.

FSAR 15.9.1 Asymmetric Steam Generator Transient

FSAR Section 15.9.1.1 describes the Asymmetric Steam Generator Transient (ASGT). The transients resulting from the malfunction of one steam generator are analyzed to determine the initial margins that must be maintained by the technical specifications limiting conditions for operation (LCOs) such that in conjunction with the RPS (CPC high differential cold leg temperature reactor trip) the DNBR and Fuel Centerline Melt (CTM) limits are not exceeded.

The asymmetric steam generator transient was submitted and approved for the extended power uprate. Plant changes since the extended power uprate have been incorporated into the analysis under the 10CFR50.59 process. The analysis has been updated to account for the replacement steam generators and the NGF DNBR correlation.

FSAR Section 15.9.1.1.3.3 states that the required margin for this event is less than that required for the FSAR Section 15.4.1.4 CEA drop event. FSAR 15.9.1.1.5 states that the radiological consequences due to steam releases from the secondary system are less

severe than the consequences of the FSAR Section 15.1.2.4 inadvertent opening of an atmospheric dump valve.

The AOR changes are the revised CEA SCRAM insertion curve, a decrease in ASI from +0.3 to +0.2, and the use of the actual initial thermal margin reserved in the LCO. The ASGT analysis demonstrated that the initial margins were adequate to ensure that the ASGT event does not violate the DNBR (≥ 1.24) and Fuel CTM SAFDLs. The ASGT analysis demonstrated that the limits on maximum RCS and secondary pressure were not violated.

Attachment 3 to

W3F1-2015-0040

Final Safety Analysis Report Revised Assessment Summary

Attachment 2 contains the detailed evaluation of the Final Safety Analysis Report (FSAR) transients. This attachment summarizes the results for an easy reference for the transient impacts.

Key to Categories

1. Reactor trip does not occur or is not credited.
2. Consequences are not sensitive to the CEA drop time change, because of the slow rate of margin degradation through the time of trip, or due to obvious insensitivity of accident consequences as a function of the time of trip.
3. This event is bounded by another event that is presented in Chapter 15.
4. This event is assessed or evaluated for CEA drop time change impact.

FSAR Section	Event	Category	Evaluation Summary
6.2	Containment Analyses	1, 2	The LOCA and MSLB mass and energy releases are not impacted by the average CEA SCRAM insertion time curve change.
7.2.1.1.2.5 7.7.1.5	Core Protection Calculators and Core Operating Limit Supervisory System	1, 2	The CPCS pressurizer spray malfunction event is not impacted by the revised CEA SCRAM insertion curve. The CPCS excess load event is not impacted by the revised CEA SCRAM insertion curve. The CPCS bank CEA withdrawal event is not impacted by the revised CEA SCRAM insertion curve.
7.7.1.9	Reactor Power Cutback System	1, 2	The CPCS reactor power cutback event is not impacted by the revised CEA SCRAM insertion curve.
9.3.6	BTP 5-4 Natural Circulation Cooldown Analysis	2	Loss of Normal Feedwater Flow event demonstrated the event characteristics would not be impact by the revised CEA drop time.
9.5	Appendix R	2	Loss of Normal Feedwater Flow event demonstrated the event characteristics would not be impact by the revised CEA drop time.
15.1.1.1	Decrease in Feedwater Temperature	3	This event is bounded by the Increase in Main Steam Flow events (UFSAR Sections 15.1.1.3 / 15.1.2.3) which are evaluated separately.
15.1.1.2	Increase in Feedwater Flow	3	This event is bounded by the Increase in Main Steam Flow events (UFSAR Sections 15.1.1.3 / 15.1.2.3), which are evaluated

FSAR Section	Event	Category	Evaluation Summary
			separately.
15.1.1.3	Increase Main Steam Flow	2	The excess load events are bounded by the current AOR.
15.1.1.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve	2	The IOADV events are not impacted by the revised CEA SCRAM insertion curve.
15.1.2.1	Decrease in Feedwater Temperature	3	This event is bounded by the Increase in Main Steam Flow events (UFSAR Sections 15.1.1.3 / 15.1.2.3), which are evaluated separately.
15.1.2.2	Increase in Feedwater Flow	3	This event is bounded by the Increase in Main Steam Flow events (UFSAR Sections 15.1.1.3 / 15.1.2.3), which are evaluated separately.
15.1.2.3	Increased Main Steam Flow	4	The increased main steam flow event has been evaluated with respect to the current AOR and the results remain with the acceptance criteria.
15.1.2.4	Inadvertent Opening of a Steam Generator Atmospheric Dump Valve	4	The inadvertent opening of an atmospheric dump valve event has been evaluated with respect to the current AOR and the results remain with the acceptance criteria.
15.1.3.1	Steam System Piping Failures Post-Trip Return-To-Power	4	The steam system piping failures post-trip return-to-power events are bounded by the current AORs.
15.1.3.2	Steam System Piping Failures Inside and Outside Containment Modes 3 and 4 with All CEAs on the Bottom	2	The steam system piping failures inside and outside containment Modes 3 and 4 with all CEAs on the bottom events are not impacted by the revised CEA SCRAM insertion curve.
15.1.3.3	Steam System Piping Failures: Pre-Trip Power Excursion Analysis	4	The steam system piping failures pre-trip power excursion has been evaluated with respect to the current AOR and the results remain with the acceptance criteria.
15.2.1.1	Loss of External Load	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3), which are evaluated separately.

FSAR Section	Event	Category	Evaluation Summary
15.2.1.2	Turbine Trip	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3), which are evaluated separately.
15.2.1.3	Loss of Condenser Vacuum	4	The LOCV event results demonstrate that the peak primary and secondary pressures remain below the acceptance criteria with only a minimal increase in peak primary pressure.
15.2.1.4	Loss of Normal AC Power	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3) and Total Loss of Forced Reactor Coolant Flow event (Section 15.3.2.1), which are evaluated separately.
15.2.1.5	Steam Pressure Regulator Failure	3	Impact of the revised CEA SCRAM insertion curve is accounted for in the evaluation of the LOCV events.
15.2.2.1	Loss of External Load	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3), which are evaluated separately.
15.2.2.2	Turbine Trip	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3), which are evaluated separately.
15.2.2.3	Loss of Condenser Vacuum	3	This event is bounded by the FSAR Section 15.2.1.3 Loss of Condenser Vacuum.
15.2.2.4	Loss of Normal AC Power	3	This event is bounded by the Loss of Condenser Vacuum events (UFSAR Sections 15.2.1.3 / 15.2.2.3) and Total Loss of Forced Reactor Coolant Flow event (Section 15.3.2.1), which are evaluated separately.
15.2.2.5	Loss of Normal Feedwater Flow	4	The impact of the loss of feedwater flow event has been evaluated and remains below the limits defined in the current AORs.
15.2.3.1	Feedwater System Pipe Breaks	4	The impact on the feedwater system pipe break events have been evaluated and remains well below the limits defined in the current AORs.

FSAR Section	Event	Category	Evaluation Summary
15.2.3.2	Loss of Normal Feedwater Flow with an Active Failure in the Steam Bypass System	4	The impact of the loss of feedwater flow event has been evaluated and remains below the limits defined in the current AORs.
15.3.1.1	Partial Loss of Forced Reactor Coolant Flow	3	This event is bounded by the Total Loss of Forced Reactor Coolant Flow event (UFSAR Section 15.3.2.1), which is evaluated separately.
15.3.2.1	Total Loss of Forced Reactor Coolant Flow	4	The loss of flow event has been evaluated with respect to the current AOR and the results remain with the acceptance criteria.
15.3.2.2	Partial Loss of Forced Reactor Coolant Flow	3	This event is bounded by the Total Loss of Forced Reactor Coolant Flow event (UFSAR Section 15.3.2.1), which is evaluated separately.
15.3.3.1	Single Reactor Coolant Pump (RCP) Shaft Seizure / Sheared Shaft	4	The single shaft seizure / sheared shaft event has been evaluated with respect to the current AOR and the results remain with the acceptance criteria.
15.4.1.1	Uncontrolled CEA Withdrawal from a Subcritical	4	The uncontrolled CEA bank withdrawal from a subcritical condition has been evaluated and continues to have margin to the limits defined in the current AORs.
15.4.1.2	Uncontrolled CEA Withdrawal at Low Power	4	The uncontrolled CEA bank withdrawal at low power has been evaluated and continues to have margin to the limits defined in the current AORs.
15.4.1.3	Uncontrolled CEA Withdrawal at Power	2	The CEA bank withdrawal is bounded by the current AOR.
15.4.1.4	CEA Misoperation	1	The single and subgroup (CEACs inoperable) CEA Drop events are not impacted by the revised CEA SCRAM insertion curve. The Single CEAW event is bounded by the current AOR.
15.4.1.5	CVCS Malfunction (Inadvertent Boron Dilution)	1, 3	The CVCS Malfunction (IBD) is not impacted by the revised CEA SCRAM insertion curve.
15.4.1.6	Startup of an Inactive Reactor coolant System Pump	1	The startup of an inactive reactor coolant system pump event is not impacted by the revised CEA SCRAM insertion curve.
15.4.1.7	Uncontrolled CEA Withdrawal from a Subcritical Condition –	2	The uncontrolled CEA withdrawal from a subcritical condition in Operating Modes 3, 4, and 5 with all CEAs on the bottom event is not

FSAR Section	Event	Category	Evaluation Summary
	Mode 3, 4, 5 With All CEAS on the Bottom		impacted by the revised CEA SCRAM insertion curve.
15.4.3.1	Inadvertent Loading of a Fuel Assembly into the Improper Position	1	The inadvertent loading of a fuel assembly into the improper position event is not impacted by the revised CEA SCRAM insertion curve.
15.4.3.2	CEA Ejection	4	The impact of the CEA Ejection events have been evaluated and remains below the limits defined by the current AORs.
15.5.1.1	CVCS Malfunction	2	The CVCS Malfunction events (pressurizer fill) for HFP are not impacted by the revised CEA SCRAM insertion curve.
15.5.1.2	Inadvertent Operation of the ECCS During Power Operation	1	The inadvertent operation of the ECCS during power operation event is not impacted by the revised CEA SCRAM insertion curve.
15.5.2.1	CVCS Malfunction	2	The CVCS Malfunction events (pressurizer fill) for HFP are not impacted by the revised CEA SCRAM insertion curve.
15.6.3.1	Primary Sample or Instrument Line Break	2	The Primary Sample or Instrument Line Break (primary mass releases) event is not impacted by the revised CEA SCRAM insertion curve.
15.6.3.2	Steam Generator Tube Rupture	2	The impact on the steam generator tube rupture events have been evaluated and remains well below the limits defined by the current AORs.
15.6.3.3	Loss of Coolant Accident	1, 2	The large break LOCA and Post-LOCA Long Term cooling do not model CEA insertion. The small break LOCA CEA insertion reactivity curve bounds the revised CEA drop time curve.
15.6.3.4	Inadvertent Opening of a Pressurizer Safety Valve	3	The inadvertent opening of a pressurizer safety valve event are bounded by the results of the Small Break LOCA.
15.7.3.3	Postulated Radioactive Releases Due to Liquid Containing Tank Failures	1	The postulated radioactive release due to liquid containing tank failure events are not impacted by the revised CEA SCRAM Insertion curve.
15.7.3.4	Design Basis Fuel Handling Accidents	1	The design basis fuel handling accidents are not impacted by the revised CEA SCRAM insertion curve.

FSAR Section	Event	Category	Evaluation Summary
15.7.3.5	Spent Fuel Cask Drop Accidents	1	The spent fuel cask drop accidents are not impacted by the revised CEA SCRAM insertion curve.
15.8	Anticipated Transients Without Scram	3	The anticipated transient without SCRAM event is not impacted by the revised CEA SCRAM insertion curve.
15.9.1	Asymmetric Steam Generator Transient	4	The ASGT analysis demonstrated that the initial margins were adequate to ensure that the ASGT event does not violate the DNBR (≥ 1.24) and Fuel CTM SAFDLs.

Attachment 4 to

W3F1-2015-0040

Average Control Element Assembly Drop Time Basis

Technical Specification 3.1.3.4 Average Drop Time Confirmation

The Technical Specification (TS) 3.1.3.4 required Control Element Assembly (CEA) drop time measurement is performed at the beginning of every cycle to ensure the CEAs adequately trip when required. The current TS 3.1.3.4 CEA drop time limits were approved by the NRC in TS Amendment 58 [Reference 2]. The average drop time limit submittal [Reference 1] justified that negative reactivity insertion for any reasonable distribution of CEAs is more directly correlated to the average CEA insertion rather than by the slowest. This information is intended to validate that the average CEA drop time methodology remains applicable to the new CEA drop times.

The reactivity worth of a CEA is a function of the power or neutron flux environment surrounding the CEA. During a reactor trip, the faster CEAs will be in higher flux regions sooner and will therefore make a greater relative contribution to the net negative reactivity insertion than the slower CEAs. Therefore, the negative reactivity insertion for any reasonable distribution of CEAs is more directly correlated to, and can be represented by, the average CEA insertion.

The NRC in TS Amendment 58 [Reference 2] found the use of the average CEA drop time acceptable with the following conditions.

(1) Any fuel management change that significantly affects the core wide axial or radial power profiles, such as axial blankets or ultra-low leakage fuel management, may necessitate re-verification of the average CEA drop time analysis.

(2) Changes that would significantly affect the CEA drop time distribution, such as changes to the CEDM circuits, large increases in the core flow pressure drop, changes in the total drop weight of the CEAs or changes in the location of the CEAs, may also require re-verification of the average CEA drop time concept.

For condition (1), the concern is related to the possibility of CEAs in peripheral assemblies inserting faster than the average, but not contributing as much to the overall reactivity inserted, due to the very low power in peripheral assemblies in such a fuel management. Waterford 3 has 87 full length CEAs. The CEAs that are above locations with one assembly corner touching the shroud tend to be fresh fuel, which have relatively high powers. Thus, this condition continues to be met.

For condition (2), the implementation of Next Generator Fuel (NGF) and the replacement Steam Generators (SGs) have caused core flow changes, but the overall CEA spatial distribution remains consistent (refer to Figure 2). Figure 1 shows the CEA core locations and Figure 2 shows the individual CEA drop times for Cycle 15 through Cycle 20. The overall drop time has increased (most likely due to the increased core pressure drop and associated increased flow rate through the guide tubes) but the distribution remains consistent for each cycle.

The NRC in TS Amendment 58 [Reference 2] also made the following statement:

> However, if the distance between the fastest and slowest CEAs becomes too large or the distribution of CEAs deviates significantly from that modeled by CE in this study, then the average CEA position (window shade) may not be representative of the time dependent reactivity insertion.

This statement applies to the individual CEA drop time which is requested to be changed from 3.2 seconds to 3.5 seconds. The 3D HERMITE analyses which support the CEA average drop time submittal [Reference 1] showed that as CEA time spread increased the rate of power reduction also increased provided that the average drop time remained the same. The main concern would be an increase in individual CEA times all in one section of the core, the CEA distribution provided in Figure 2 shows this spatial distribution remains random. For the recent CEA drops, the CEAs near the center of the core may be falling slightly faster which would be a benefit for core power reduction.

The average CEA drop time methodology proposed in Reference 1 and approved by the NRC in Reference 2 still remains applicable to Waterford 3 with the increased average and maximum CEA drop time values.

REFERENCES:

1. Waterford 3 Letter W3P89-3094, Technical Specification Change Request, Average Control Element Assembly Drop Time, August 14, 1989.
2. NRC Waterford 3 Technical Specification Amendment 58, Average Control Element Assembly Drop Time, October 31, 1989 [ADAMS Accession Number ML021760257].

Figure 1. CEA Core Locations

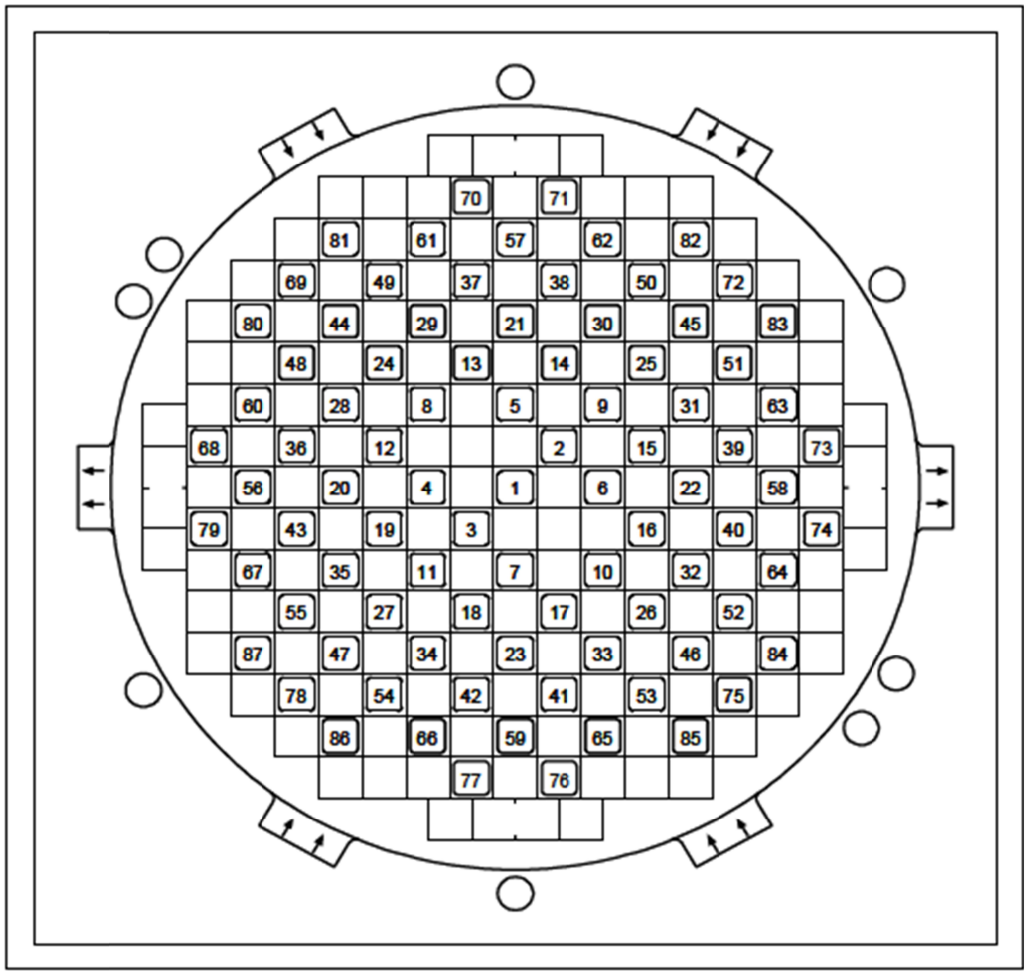
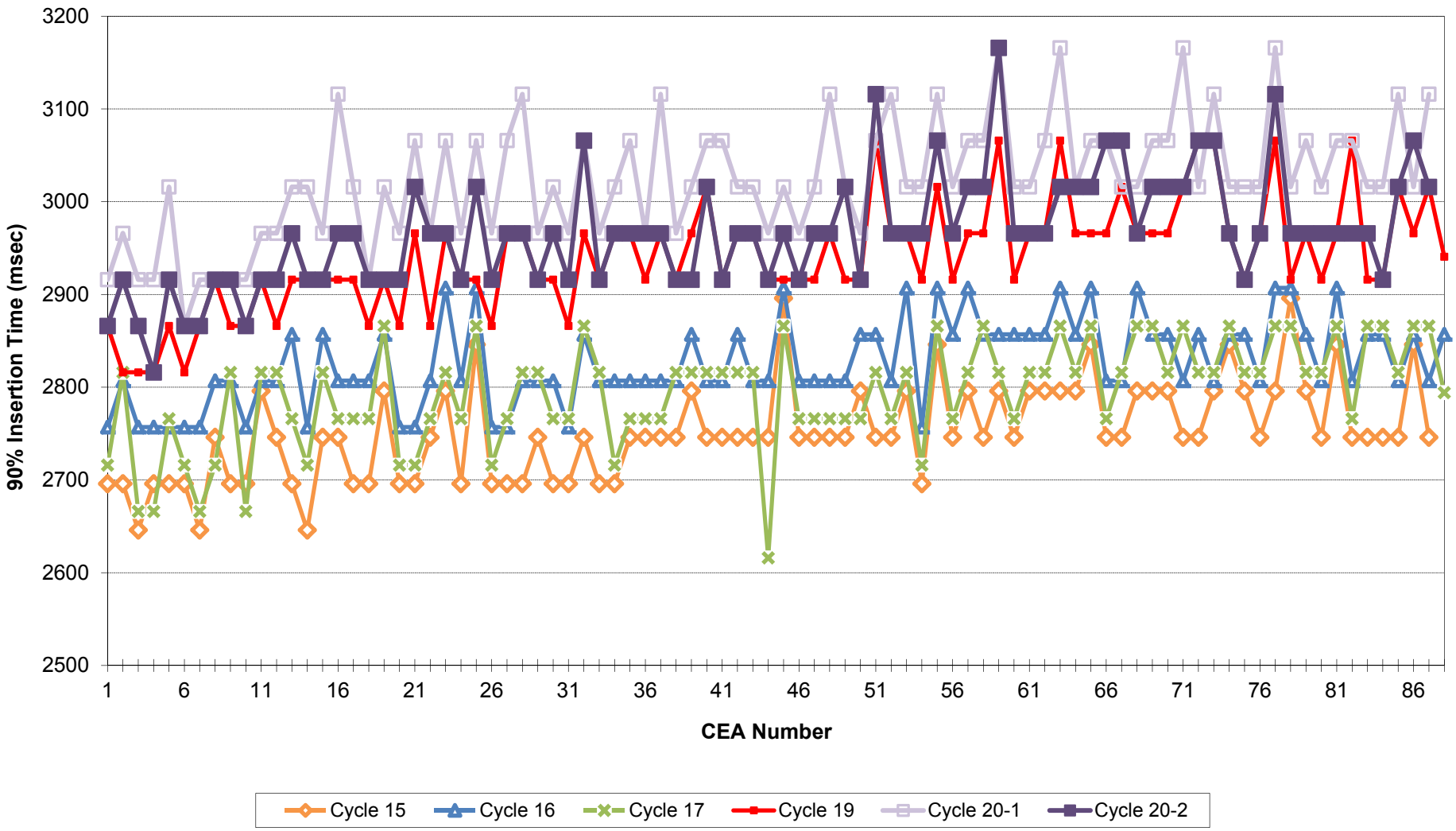


Figure 2. Cycle 15 through Cycle 20 CEA Insertion Times



Attachment 5 to

W3F1-2015-0040

Revised (Markup) Technical Specification Page

(1 Page)

REACTIVITY CONTROL SYSTEMS

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The arithmetic average of the CEA drop times of all CEAs from a fully withdrawn position, shall be less than or equal to 3.2~~3.0~~ seconds; and the individual CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.5~~3.2~~ seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with:

a. Tavg greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With either the average CEA drop time or any individual CEA drop time determined to exceed the above limits, restore the CEA drop time to within the above limits before proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At each refueling outage.

Attachment 6 to

W3F1-2015-0040

Revised (Clean) Technical Specification Page

(1 Page)

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The arithmetic average of the CEA drop times of all CEAs from a fully withdrawn position, shall be less than or equal to 3.2 seconds; and the individual CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.5 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches the 90% insertion position with:

a. Tavg greater than or equal to 520°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With either the average CEA drop time or any individual CEA drop time determined to exceed the above limits, restore the CEA drop time to within the above limits before proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal and reinstallation of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At each refueling outage.

Attachment 7 to

W3F1-2015-0040

List of Regulatory Commitments

List of Regulatory Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The Cycle 21 CEA drop time surveillance data will be provided to the NRC to confirm the conclusion of no further degradation.	X		Within 60 days of surveillance completion.
The Cycle 21 CEA drop time data will be analyzed to validate CEA insertion curve remains within the analysis requirements.	X		Within 60 days of surveillance completion.
The limiting accident events will be evaluated for CEA holding coil decay time of 0.8 seconds.	X		07/31/15
The radial power fall-off curve limits shall be verified each cycle as part of the Westinghouse reload analysis methodology until a new licensing basis long term fuel methodology is approved Waterford 3. Upon NRC approval of a new long term fuel evaluation model and associated methods that explicitly account for thermal conductivity degradation (TCD) that is applicable to Waterford Unit 3 design, Entergy will, within 6 months: a. Demonstrate that Waterford Unit 3 safety analysis remain conservatively bounded in licensing basis analyses when compared to the NRC-approved new long term fuel evaluation model that is applicable to Waterford Unit 3 design, and/or b. Provide a schedule for re-analysis using the NRC-approved new long term fuel evaluation model that is applicable to Waterford 3 design for any affected licensing basis analyses.		X	Commitment description contains time requirement.

Attachment 8 to

W3F1-2015-0040

Fuel Thermal Conductivity Degradation Evaluation Affidavit

Affidavit to Withhold from Public Disclosure
Proprietary Information
Under 10 CFR 2.390

As Attachment 9 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Westinghouse

Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com
Proj letter: LTR-SCC-15-028

CAW-15-4219

June 26, 2015

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: CE-15-284, Attachment 2, "Fuel Management Adjustment to Radial Fall-off to Reserve Margin for Thermal Conductivity Degradation" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-15-4219 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

The subject document was prepared and classified as Westinghouse Proprietary Class 2. Westinghouse requests that the document be considered proprietary in its entirety. As such, a non-proprietary version will not be issued.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Entergy Operations, Inc.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-15-4219, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

James A. Gresham, Manager
Regulatory Compliance

CAW-15-4219
June 26, 2015

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Henry A. Sepp, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Henry A. Sepp, Director
CRE-Systems and Components Engineering

(1) I am Director, CRE-Systems and Components Engineering, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is contained in CE-15-284, Attachment 2, "Fuel Management Adjustment to Radial Fall-off to Reserve Margin for Thermal Conductivity Degradation" (Proprietary), for submittal to the Commission, being transmitted by Entergy Operations, Inc. letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with Fuel Performance, Safety Analysis and the associated Thermal Conductivity Degradation methodologies and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Perform Reload Fuel and Safety analyses

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of performing reload fuel and safety analyses

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent saycth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith is the proprietary version of a document furnished to the NRC associated with Fuel Performance, Safety Analysis and the associated Thermal Conductivity Degradation methodologies and may be used only for that purpose. The document is to be considered proprietary in its entirety.

COPYRIGHT NOTICE